Report No. IDO-28598
Contract AT(10-1)-880

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ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

FINAL SUMMARY REPORT OF THE

GAS-COOLED REACTOR EXPERIMENT-I

October 1963

GENERAL TIRE

AEROJET-GENERAL NUCLEONICS

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Report No. IDO-28598
UC-80, "Reactor Technology"
TID-4500 (22nd Edition)

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Approved by:

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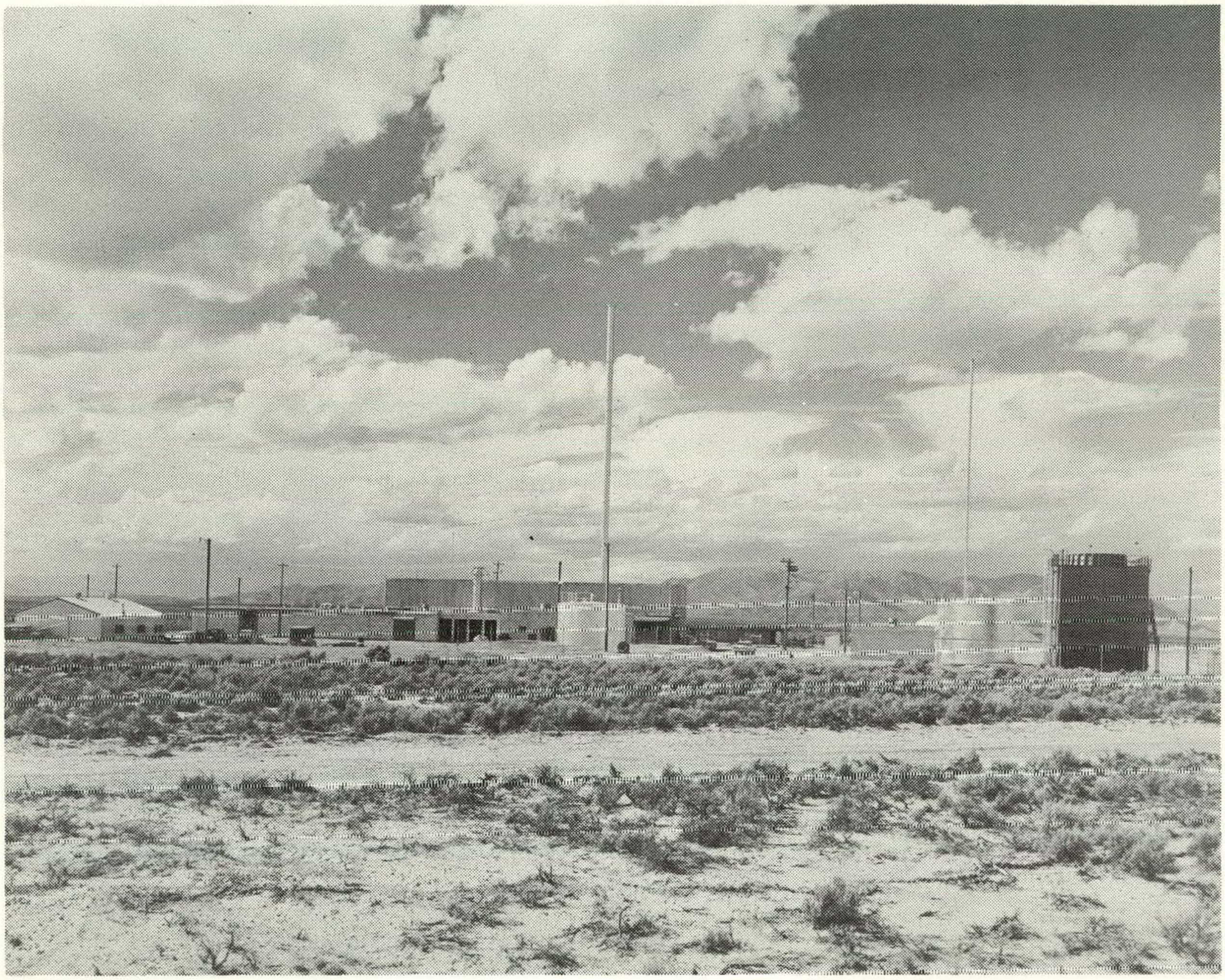
AEROJET-GENERAL NUCLEONICS
A SUBSIDIARY OF AEROJET-GENERAL CORPORATION

GENERAL TIRE

Printed in the USA. Price $3.00
Available from the Office of
Technical Services, Department
of Commerce, Washington 25, D.C.



GENERAL VIEW OF THE GCRE TEST FACILITY AT THE NRTS, IDAHO

Report No. IDO-28598

ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

FINAL SUMMARY REPORT OF THE GAS-COOLED REACTOR EXPERIMENT-I*

ABSTRACT

This report summarizes the Gas-Cooled Reactor Experiment-I (GCRE-I), conducted at the National Reactor Testing Station, Idaho, by Aerojet-General Nucleonics as a major project under the Army Gas-Cooled Reactor Systems Program. Initial criticality of the GCRE-I reactor was achieved on 23 February 1960 and testing and evaluation were continued until April 1961.

The mission of the GCRE-I was the generation of neutronic and engineering data in support of the design and development of a nuclear energy source for a demonstration low-power gas-cooled, water-moderated power plant.

The GCRE reactor was a heterogeneous, fully-enriched uranium-fueled unit operated in a pool of demineralized water which served as the moderator and primary radiological shield. Fuel elements were contained in a tube bundle, or calandria, which separated the fuel from the water moderator and provided passages for the gas coolant.

Test operations and experimental evaluations were conducted with two core loadings of different design (one consisting of dispersion plate-type fuel elements and the other of pin-type elements) and two calandrias of similar design (one fabricated of aluminum and the other of stainless steel).

The report includes a description of the facility and the test reactor, a full account of the conduct and results of the experimental program and a discussion of the operating organization and philosophy.

A bibliography of reports covering the GCRE program is presented.

*Published by Aerojet-General Nucleonics, San Ramon, California. Work described was performed by AGN under Contract AT(10-1)-880 with the U. S. Atomic Energy Commission.

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ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

FINAL SUMMARY REPORT OF THE
GAS-COOLED REACTOR EXPERIMENT-I

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ARMY GAS-COOLED REACTOR SYSTEMS PROGRAM

FINAL SUMMARY REPORT OF THE GAS-COOLED REACTOR EXPERIMENT-I

I. INTRODUCTION

The Gas-Cooled Reactor Experiment-I (GCRE-I)* was conceived in mid-1957 as a major project under the Army Gas-Cooled Reactor Systems Program (AGCRSP). Earlier studies by Oak Ridge School of Reactor Technology (ORSORT), Sanderson-Porter Company (Refs. 1,2)** and Aerojet (Refs. 3,4,5,6,7,8) had indicated that the development of a mobile, nuclear power plant for military applications was feasible. These studies had specified a heterogeneous, gas-cooled, water-moderated reactor for such a plant. The goal of the GCRE program was the test operation of a developmental model of that reactor to provide engineering data for the design of a reactor for a demonstration power plant, the Portable Gas-Cooled Reactor (PGCR).

The objectives of the GCRE-I program were:

1) Design and construction of a test facility at the National Reactor Testing Station (NRTS), Idaho, suitable for test operation and evaluation of the experimental reactor. (In this report the test facility is referred to as the GCRE.)

2) Design and fabrication of a test reactor and associated controls for installation in the GCRE. (In this report the test reactor is referred to as the GCRE-I.)

3) Test operation of the GCRE-I to evaluate reactor neutronic characteristics, fuel element performance, and closed-cycle gas-cooled system characteristics.

* See Appendix A for List of Abbreviations

**Numbers in parentheses identify references listed at the end of this report.

The design of the GCRE-I was begun in 1957 by Aerojet under a contract with the U. S. Atomic Energy Commission (USAEC) administered by the Idaho Operations Office. Design of the GCRE began shortly thereafter and construction was initiated in the spring of 1958. Fabrication of the GCRE-I was completed in September 1959, and construction of the GCRE was completed the following month.

In the meantime, in June 1959, the Department of Defense (DOD) and the USAEC reoriented the basic missions of the AGCRSP. The PGCR was eliminated from the program and firm Military Characteristics, which were, in effect, performance specifications, were established for the mobile, nuclear power plant (see Appendix B). The primary goal of the revised program was the demonstration operation of a prototype plant, designated ML-1, by May 1962.

An evaluation of the ability of the GCRE-I design to satisfy the ML-1 Military Characteristics revealed the following:

1) The GCRE-I was too large to satisfy the weight and size specifications. The ML-1 reactor would have to be more compact in order that the reactor skid, including the required shielding, could be packaged in conformance with the weight and envelope specifications.

2) The original GCRE-I dispersion plate-type fuel element concept was unsuitable for use in the more compact ML-1 reactor; the higher fuel loadings required for the ML-1 would force abandonment of the plate-type concept in favor of a design capable of higher fuel concentrations.

3) The GCRE-I operating conditions did not satisfy the ML-1 system requirements; temperatures were satisfactory but the coolant flow rate, system pressure, and reactor power were somewhat lower than required for testing at ML-1 operating conditions.

Despite these shortcomings, however, the evaluation concluded that test operation of the GCRE-I could contribute significantly to the ML-1 development program in the following areas:

1) Development of neutronic data relating to highly heterogeneous, water-moderated cores - The evolution of analytical procedures to predict the neutronic behavior of such cores had proven difficult and the availability of experimental data which could be used to check and normalize the calculations would be very valuable in the analysis and design of the ML-1 core.

2) Statistical performance and life testing of fuel elements of the ML-1 prototype design - Such testing would permit the assignment of significantly higher confidence levels to the conclusions relating to fuel element performance than would be possible from in-pile tests alone.

3) Evaluation of system performance characteristics - In view of the limited amount of data available on closed-cycle gas power plants, the behavior of the GCRE system would provide significant information on

system transfer functions, potential instabilities, etc., which could be extrapolated with reasonable confidence to ML-1 operating conditions.

4) Development of reactor engineering and performance data - Direct experimental evaluation of such problems as afterheat generation rates, photoneutron production, local boiling of moderator, flanged closure integrity and materials corrosion would increase the level of confidence in the ML-1 analytical and laboratory test programs and might reveal unanticipated problem areas. This was particularly important since the schedule requirements and funding available for the ML-1 program did not permit a component test program.

On the basis of this evaluation, a GCRE-I experimental program was developed which would directly support the ML-1 development effort. Although the GCRE-I operation was seriously hampered by mechanical malfunctions, the test program contributed significantly to the ML-1 development effort. Considerable information was collected and, although the timing was not optimum (GCRE-I data, in general, was not available until late in the final design phase of the ML-1), a meaningful contribution to the understanding of gas-cooled closed-cycle nuclear power systems was realized.

This report summarizes the entire GCRE program. Included is a description of the GCRE facility and the GCRE-I test reactor, a discussion of the experimental program and its results and a presentation of the operating organization provided and the philosophy employed in the execution of the work. A bibliography of reports covering the GCRE program is presented in Appendix C.

II. SUMMARY

The mission of the GCRE test facility was to perform the operational testing and evaluation of a test reactor in support of the development of the ML-1 gas-cooled nuclear power plant under the Army Gas-Cooled Reactor Systems Program. The facility was designed by the Architect-Engineer Division of Aerojet-General Corporation under contract with the USAEC (AT(10-1)-880) and constructed (in two phases) by Farnsworth and Chambers, Inc. (AT(10-1)-945) and Anderson-Burke Co. (AT(10-1)-999) under direct contract with the Idaho Operations Office of the USAEC. The facility construction was completed on 22 September 1959.

The GCRE-I test reactor was designed, fabricated and installed by Aerojet-General Nucleonics (AGN), a subsidiary of the Aerojet-General Corporation , under Contract (AT(10-1)-880) with the Idaho Operations Office of the USAEC. The initial criticality of the reactor was achieved on 23 February 1960 and operation at design full power was initiated on 22 July 1960. Test operations and experimental evaluations were conducted with two core loadings of different design (one consisted of dispersion plate-type fuel elements, designated IZ, and the other of pin-type elements, designated IB-2L) and two calandrias of similar design (one fabricated of aluminum and the other of stainless steel). A total of 2989 Mw-hr was accumulated on the reactor, including approximately 1000 hr at full power. The reactor was shut down on 6 April 1961 because of a leak in the stainless steel calandria. The subsequent diagnostic program resulted in the decision in January 1962 to deactivate the GCRE project. The deactivation work was essentially completed by 1 July 1962.

A summary of the GCRE-I reactor operating conditions and neutronic and mechanical characteristics is given in Table II-1. A tabulation of the major experiments conducted at the GCRE, along with the significant results of the experiments, is presented in Table II-2.

TABLE II-1

GCRE-I REACTOR DESIGN AND OPERATING CHARACTERISTICS

	Plate-Type (IZ) Fuel Elements-Aluminum Calandria	Pin-Type (IB-2L) Fuel Elements-Stainless Steel Calandria
1. Operating Characteristics		
Power, Mw(t), design maximum	2.2	2.2
Power density, kw/ft^3	300	360
Inlet pressure, psia, nominal	200	200
Inlet temperature, ^oF, nominal	800	800
Outlet pressure, psia, nominal	187	182
Outlet temperature, ^oF, nominal	1200	1200
Fuel element temp , ^oF, maximum (hot spot temperature)	1650	1750
Coolant	N_2 with 0.5% O_2	N_2 with 0.5% O_2
Coolant flow rate, nominal, lb/hr	63,000	63,000
2. Reactor Core		
Number of fuel elements (operating)	61	56
Critical mass, kg (cold, clean)	13.8	36.3
Operating loading, kg	19.2	44.7
Reflector		
top	2 in. water and 3½ in. aluminum	2 in. water and 2½ in. stainless steel
bottom	3½ in. aluminum	6 in. water and 3½ in. stainless steel
radial	4 in. lead and water	4 in. lead and water
Neutron flux		
thermal, average, n/cm^2-sec	3×10^{12}	3.5×10^{12}
fast, average, n/cm^2-sec	1.9×10^{13}	1.3×10^{13}
Reactivity coefficients		
Moderator $\Delta K/K$-^oF (80-100^oF)	----	$+0.9 \times 10^{-4}$
Coolant temperature, $\Delta K/K$-^oF (100-900^oF)	----	$+0.34 \times 10^{-6}$

TABLE II-1 - Continued

	Plate-Type (IZ) Fuel Elements-Aluminum Calandria	Pin-Type (IB-2L) Fuel Elements-Stainless Steel Calandria
Coolant pressure, $\Delta K/K$-psi, (0-200 psi)	----	-3.0×10^{-6}
3. Control Rods	Identical for both core loadings	
Number, total	12	
Number by type		
Safety	2	
Shutdown	6	
Shim	2	
Setback	1	
Regulating (fine)	1	
Absorber material by rod type:		
Safety	Tungsten, cadmium	
Shutdown	Tungsten, indium, cadmium	
Setback, shim and regulating	Tungsten	
Reactivity worth by rod type, % $\Delta K/K$, unshadowed		
Safety (two)	2	
Shutdown (six)	6	
Shim, setback and regulating	2.5	
Total	10.5	
4. Calandria		
Pressure tubes		
Number	73	73
Length	37	37
Outside diameter, in.	1.875	1.800
Wall thickness, in.	0.058	0.020
Material	Aluminum	AISI Type 321 Stainless Steel

TABLE II-1 - Continued

	Plate-Type (IZ) Fuel Elements-Aluminum Calandria	Pin-Type (IB-2L) Fuel Elements-Stainless Steel Calandria
Tube sheet		
Material	6061-T6 Aluminum	AISI Type 304 Stainless Steel
Diameter, in.	40	40
Thickness, in.	3	3.5
3. Plenums		
Height, in.	16	16
Outside diameter, in.	28	28
Wall thickness, in.	0.375	0.375
Material	AISI Type 304 Stainless Steel	AISI Type 304 Stainless Steel

TABLE II-2

SUMMARY OF GCRE-I EXPERIMENTAL PROGRAM

Experiment	Purpose	Results
INITIAL (WET) CRITICAL (9200)*	Define critical loading with plate-type (IZ) elements in flooded core	36 plate-type elements required for criticality
FULL FLOODED CORE (9201)	Demonstrate subcriticality of fully loaded core in flooded condition with two control rods inoperable	61 element plate-type core subcritical flooded with two strongest rods out
REACTOR DRYING (9204)	Demonstrate effectiveness of drying procedures	Loop dew-point minus 40°F after 84 hr
DRY CRITICAL (9202)	Define critical loading with plate-type elements in operating (dry) condition	59 plate-type elements required for criticality
OPERATING CORE LOADING (9203)	Define operating core loading	61 plate-type elements; $k_{excess} = \sim 0.7\% \Delta K/K$
PRELIMINARY INSTRUMENT CALIBRATION (9205)	Establish actual reactor operating power	Satisfactorily completed
CONTROL ROD CALIBRATION (9206)	Check procedures, calibrate fine rod, determine other rod worths	Completed; procedures generally acceptable
MODERATOR TEMPERATURE (9208)	Determine moderator temperature coefficient of reactivity	$+ 0.9 \times 10^{-4} \Delta K/K\text{-}^{\circ}F$
FLUX MAPPING (9209)	Check procedures for radial and axial flux and power distribution determinations	Completed; procedure and equipment generally satisfactory
INITIAL (WET) CRITICAL (9500)	Define critical loading with plate-type elements in flooded stainless steel calandria	52 plate-type elements required for criticality
FULL FLOODED CORE (9501)	Same as 9201 except with stainless steel calandria	72 element plate-type core subcritical with two strongest rods out

*Refers to the standard operating procedure (ANSOP) which specified experimental procedure.

TABLE II-2 - Continued

Experiment	Purpose	Results
DRY CRITICAL (9502)	Define critical loading of plate-type elements in stainless steel calandria in operating (dry) condition	66 plate-type elements required for criticality
OPERATING CORE LOADING (9503)	Define operating core loading in stainless steel calandria	71 plate-type elements
FLUX MAPPING (IB-9Ø)*	Define reactivity worth of pin-type elements	Completed; satisfactory
POWER ASCENSION (9210)	Raise reactor power to design value	1.85 Mw(t) limited by single element outlet gas temperature of 1300°F
PHOTONEUTRON EXPERIMENT (9510)	Establish magnitude of photoneutron source	Unsuccessful
POWER OPERATION	Evaluate plate-type elements in stainless steel calandria	750 Mw-hr accumulated on plate-type elements
INITIAL (WET) CRITICAL (9600)	Define critical loading of pin-type (IB-2L) elements	45 pin-type elements required for criticality
FULL FLOODED CORE (9601)	Same as 9501 except with pin-type elements	57-element IB-2L core subcritical with two strongest rods out
DRY CRITICAL (9602)	Define critical loading of pin-type elements in operating (dry) condition	53 pin-type elements required for criticality
OPERATING CORE (9603)	Define operating core loading	56 pin-type elements; $k_{excess} = \sim 1.6\% \Delta K/K$
REACTIVITY WORTH (9606)	Calibrate fine rod and establish fuel element reactivity worths	Completed successfully
FLUX DISTRIBUTION (9609)	Determine axial and radial power and flux distributions	Completed successfully

*Refers to test element used for flux mapping.

TABLE II-2 - Continued

Experiment	Purpose	Results
MODERATOR TEMPERATURE COEFFICIENT (9608)	Determine moderator temperature coefficient of reactivity	$+ 0.9 \times 10^{-4}$ Δ K/K-^{o}F at $\sim 100^{o}$F
REACTIVITY COEFFICIENTS (9605, 9607)	Determine temperature and pressure coefficients of reactivity	Temperature: -1×10^{7} Δ K/K-^{o}F Pressure: -7×10^{7} Δ K/K-psi
POWER ASCENSION (9610)	Raise reactor power to design value (uniform orificing)	1.8 Mw(t) limited by outlet gas temperatures of 7 central elements
ROD SHADOWING (9622)	Determine effect on fuel element temperature of rod shadowing	Completed successfully; little effect noted
SHUTDOWN COOLING (9604)	Predict reactor temperature behavior in event of coolant flow stoppage	Maximum temperatures observed did not exceed normal operating values
REACTIVITY COEFFICIENTS (9630)	Same as 9605-9607	Temperature: $+ 0.34$ $(\pm 0.21) \times 10^{-6}$ $\Delta\bar{K}$/K-^{o}F Pressure: $- 3.0$ $(\pm 1.1) \times 10^{-6}$ Δ K/K-psi
PHOTONEUTRON EXPERIMENT (9629)	Same as 9510	Ratio of photoneutron power to full power = $7 (\pm 2) \times 10^{-5}$

III. THE GCRE FACILITY

The GCRE facility is located in the Army Reactor Area (ARA) of the National Reactor Testing Station (NRTS) in southeastern Idaho. The site is approximately 40 miles west of Idaho Falls and 2 miles north of U. S. Highway 20. The contiguous areas are sparsely populated and there is no agricultural activity within the NRTS boundaries. A full discussion of population density, geology, hydrology and climatology of the area is presented in the GCRE-I Hazards Summary Report (Ref. 9).

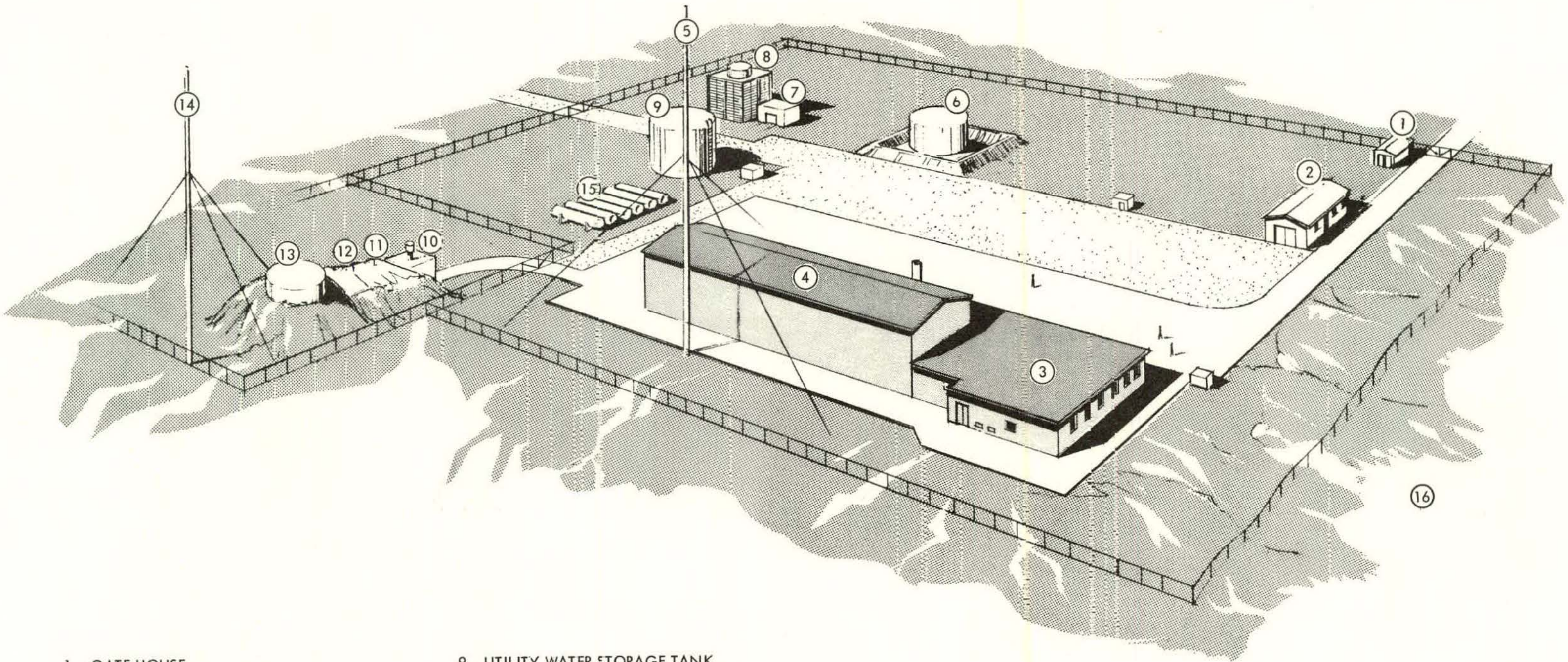
The GCRE facility consists of the Control and Test Building and several supporting and service buildings and facilities located on a 400 ft by 400 ft site surrounded by chain link security fence. The arrangement is shown in Figure III-1.

A. CONTROL AND TEST BUILDING

The Control and Test Building is a multipurpose structure approximately 200 ft by 75 ft in size. The building includes the following major areas:

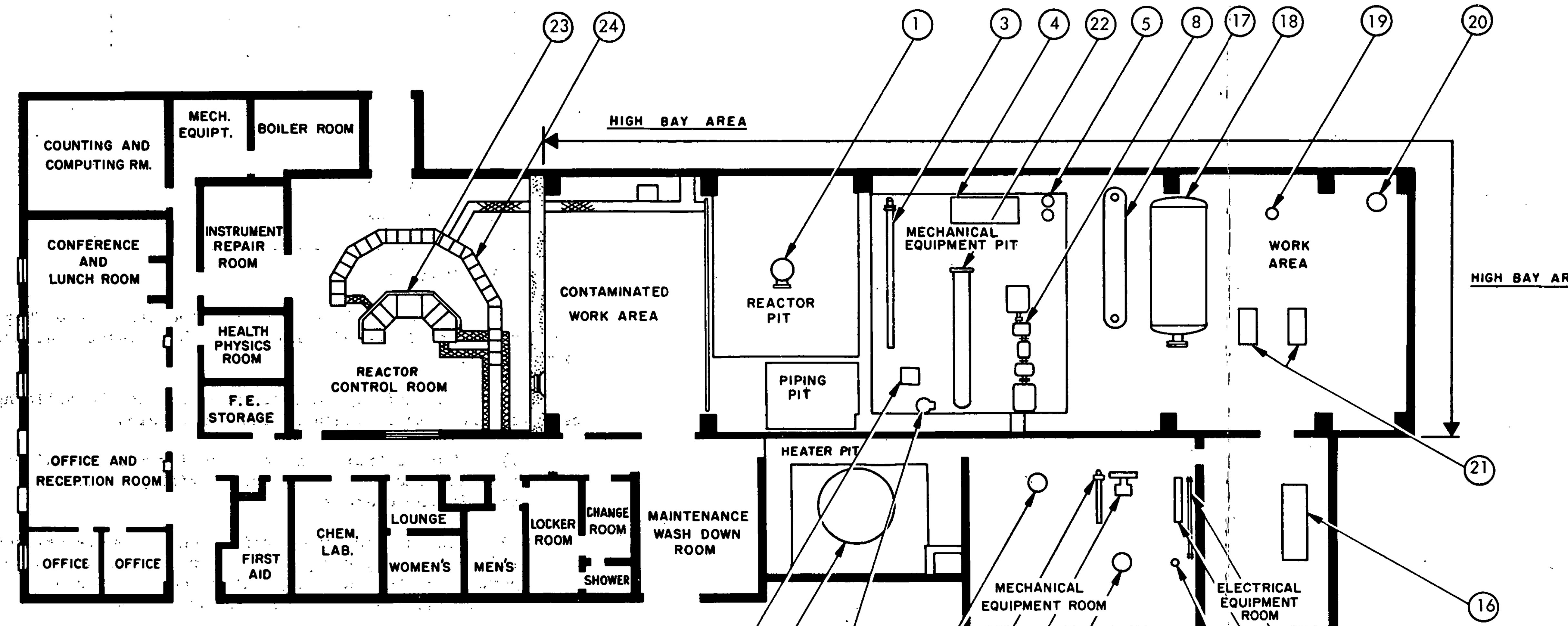
1) Offices and lunch/conference room
2) Counting and computing room
3) Instrument repair room
4) Chemistry laboratory
5) Reactor control room
6) Air conditioning and boiler room
7) Rest and ladies rooms
8) Fuel element storage and special source materials vault

All of the above are located in the one-story pumice block portion of the building on the west end and southwest side as shown in Figure III-2.



1. GATE HOUSE
2. SERVICE BUILDING
3-4. CONTROL AND TEST BUILDING
5. VENTILATION STACK
6. FUEL OIL STORAGE TANK
7. COOLING WATER PUMP HOUSE
8. COOLING TOWER
9. UTILITY WATER STORAGE TANK
10. RADIOACTIVE WASTE PUMP HOUSE
11-12. LIQUID RADIOACTIVE WASTE TANKS
13. WASTE GAS HOLDER
14. WASTE GAS STACK
15. BULK NITROGEN STORAGE
16. LEACHING PIT

FIGURE III-1. GCRE FACILITY LAYOUT



1. REACTOR
2. HEATER
3. POOL WATER HEAT EXCHANGER
4. GAS DRYER
5. GAS FILTERS
6. SHUTDOWN COOLING BLOWER
7. SHUTDOWN COOLING HEAT EXCHANGER
8. MAIN COMPRESSOR AND DRIVE
9. REACTOR FLOODING TANK
10. UTILITY AIR COOLER
11. UTILITY AIR RECEIVER
12. UTILITY AIR COMPRESSOR
13. INSTRUMENT AIR RECEIVER
14. INSTRUMENT AIR COMPRESSOR
15. INSTRUMENT AIR COOLER
16. EMERGENCY DIESEL GENERATOR
17. WATER-TO-WATER HEAT EXCHANGER
18. DEMINERALIZED WATER SURGE DRUM
19. 7 GPM DEMINERALIZER
20. 50 GPM DEMINERALIZER
21. DEMINERALIZED WATER CIRCULATING PUMPS
22. GAS-TO-WATER HEAT EXCHANGER
23. CONSOLE
24. INSTRUMENT RACKS

FIGURE III-2. GCRE CONTROL AND TEST BUILDING FLOOR PLAN

9) Maintenance wash down room

10) Heater pit

11) Mechanical equipment room

12) Electrical equipment room

The above four areas are located in the one-story pumice block portion of the building on the southeast side as shown in Figure III-2.

13) Contaminated work area

14) Reactor pit

15) Piping pit

16) Mechanical equipment pit

17) Work area

The above five areas are located in the "high-bay" portion of the building, a steel-framed structure with corrugated asbestos siding, shown in Figure III-2.

The utility services provided in the building are conventional except for the special features discussed below:

1) The control room and counting room are air-conditioned to provide a controlled temperature and humidity environment for the instrumentation in these areas.

2) The heating and ventilating system is arranged and balanced to maintain a slightly positive air pressure in the office/control room portion of the building with respect to the high-bay and associated mechanical and electrical support areas.

3) Incandescent (rather than fluorescent) lights are installed in the control room and counting room to minimize electromagnetic interference with the sensitive instruments in these areas.

4) Battery-powered emergency lights are installed at strategic locations throughout the building.

5) An induced draft exhaust system is installed in the mechanical equipment pit to discharge any contaminated gas leakage to a 150 ft high stack. This system has a capacity of 22,000 cfm.

6) An integrated emergency power system is provided to assure the continued operation of critical equipment and instrumentation in the event of a power failure. This system is supplied by a 300-kw diesel-powered generator.

7) A 15-ton capacity traveling bridge crane is provided in the high-bay area.

8) An electrically-powered mobile work platform spans the reactor pit at near floor level. This device permits access for remote servicing of equipment in the pit.

B. SUPPORTING FACILITIES

The facilities provided to support the test operation of the GCRE-I reactor are discussed below. The numbers in parentheses refer to the legend of Figure III-1.

1. Utility Water System

All water required for the test site is provided by a well located under the cooling water pump house (7). In addition to the well pump, this structure also houses two cooling water circulating pumps (see discussion of cooling systems, Section III-C-2), a utility water pump, a special pump to provide additional water capacity in the event of a fire, and chlorination equipment. The chlorinated well water is stored in a 50,000 gal. tank (9) from which distribution is made by the utility water pump.

2. Radioactive Waste Handling System

Facilities for the handling and disposal of liquid and gaseous radioactive waste are provided in the northeast section of the site. Two liquid waste pumps, with associated valves and piping are housed in the radioactive waste pump house (10) and two 8 by 27 ft tanks (11, 12) are provided for storage and treatment of the liquid waste prior to release to the leaching pit (16). A 10,000-cu ft gas holder (13) is provided for the collection of gaseous radioactive waste. Potentially contaminated gas is exhausted from the gas holder to the atmosphere through a 150-ft high stack (14). Provision is made for dilution of this effluent stream with air between the gas holder and the stack (Figure III-3).

3. Process Materials Storage

High pressure tanks are provided for the bulk storage of nitrogen gas (15). This facility has a capacity of 175,000 cu ft of gas.

Fuel oil, for the main loop heater and miscellaneous heating requirements, is stored in a 42,000 gal. tank (6). Circulating pumps are provided to distribute the oil.

4. Miscellaneous Support Facilities

A gate house (1) is provided for the security guard who controls access to the area.

A service building (2) is provided for routine mechanical maintenance equipment and materials storage.

A public address and evacuation alarm system services all parts of the facility.

FIGURE III-3. GCRE RADIOACTIVE WASTE HANDLING AREA

The pump house is shown in the center; the two liquid waste tanks are immediately behind the pump house and covered with earth shielding; the top portion of the gas holder protrudes above the earth shielding in the background and the stack is visible behind the gas holder. The utility water storage tank appears in the immediate foreground.

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Fire hose storage houses, fire hydrants and a complete water distribution system are provided for fire fighting. Special fire hazard areas are equipped with Foamite (oil storage tank) or CO_2 (oil-fired heater pit) systems. Hand fire extinguishers are provided throughout the facility.

C. PROCESS EQUIPMENT

The process equipment is provided primarily to circulate coolant gas through the reactor at controlled temperature, pressure and flow conditions. This function is performed by the components of the main coolant loop; several auxiliary systems are provided to support the operation of the main loop or to permit the performance of maintenance or other non-operating functions. Flow diagrams for the GCRE process systems are presented in Appendix D.

1. Main Coolant Loop

A schematic diagram of the GCRE main loop is presented in Figure III-4. Coolant gas (99.5% N_2; 0.5% O_2) leaves the reactor (upper left) and passes through a gas-to-water heat exchanger to the suction of the circulating compressor. The compressor discharges gas through an oil-fired heater and the cycle is completed as the gas re-enters the reactor.

The heat exchanger is a shell and tube type; the gas makes a single pass through the shell side while the demineralized cooling water makes two passes through the U-shaped tubes. The unit is designed to transfer 18×10^6 Btu/hr at a gas flow of 63,000 lb/hr and a water flow of 1000 gpm (475,000 lb/hr). The design temperature conditions are:

	Gas Side	Water Side
Inlet Temperature, ^oF	1200	100
Outlet Temperature, ^oF	140	140

The circulating compressor is designed to pump 1200 cfm of nitrogen at a nominal discharge pressure of 200 psia. The compressor is driven through a geared speed increaser and eddy current speed controller by a 450 hp induction motor. An emergency 60 hp electric motor is placed in the drive system between the speed controller and the gearbox. During normal operation this unit "floats"; in case of a power failure, the speed controller is de-energized (which uncouples the 450 hp motor) and the 60 hp motor is energized from an emergency power source to provide continued circulation of coolant gas. A cooling water circuit to the speed controller provides a flow of 35 gpm of water at a pressure of 20-35 psig. The compressor is equipped with a water-buffered seal and a common lubrication system is provided for all rotating components.

The oil-fired heater is a single pass furnace capable of transferring 10×10^6 Btu/hr to the flowing gas stream. In normal operation at 17.5 lb/sec gas flow, the outlet temperature is 800^oF. The heater burns No. 2 fuel oil at a rate of about 125 lb/hr. Startup is accomplished by spark ignition of liquified petroleum gas pilot burners. A bypass line is provided

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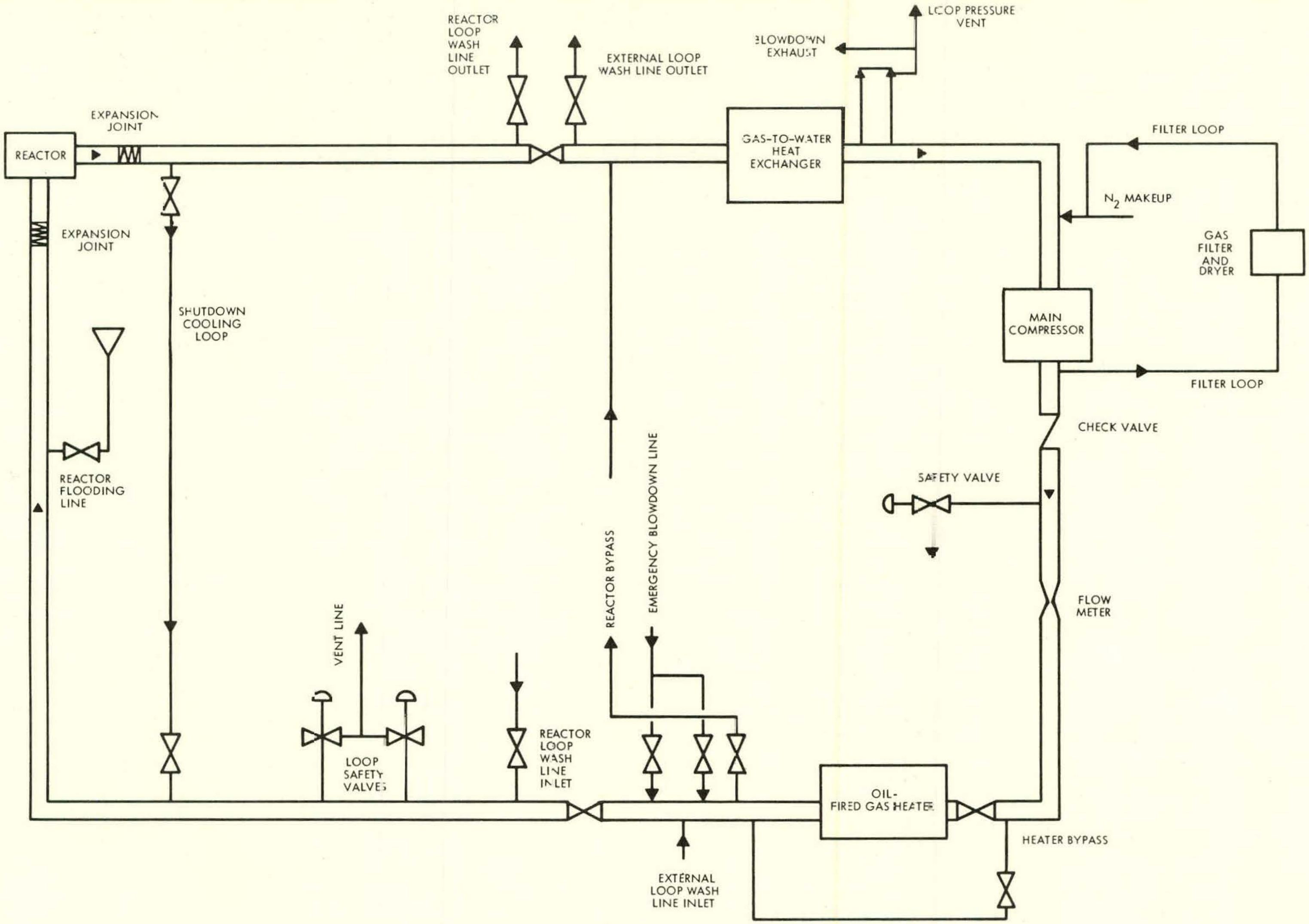

FIGURE III-4. GCRE MAIN COOLANT LOOP SCHEMATIC DIAGRAM around the heater, and valves in this line and the main line are operated automatically to control the gas outlet temperature at the desired level.

Block valves are provided between the reactor and the heat exchanger and between the heater and the reactor to permit "splitting" the loop into an internal (reactor) loop (gas flow through the shutdown cooling system) and an external loop (gas flow through the reactor bypass line). This arrangement permits operation of the loop mechanical components without flowing gas through the reactor or, conversely, maintenance of the external loop components while reactor cooling is provided by the shutdown cooling system.

Pressure, temperature and flow instrumentation is provided throughout the loop to monitor the operating parameters and provide signals for the several automatic control systems. All pertinent parameters are displayed on instrumentation in the control room. The general arrangement of the main coolant loop equipment is shown in Figures III-5, III-6 and III-7.

2. <u>Heat Removal System</u>

The primary purpose of the heat removal system is to reduce the temperature of the gas from the reactor from 1200^{o}F at the reactor outlet to approximately 140^{o}F at the compressor suction. This is accomplished in the main loop heat exchanger where the energy is transferred to water (demineralized to minimize scale formation in the 1200^{o}F end of the exchanger). The water is circulated in a closed loop (see Figure III-8) by either of two 1000-gpm pumps. Other components in the loop include a water-to-water heat exchanger and a surge drum. A make-up demineralizer system is provided to control the quality of water in the loop. In this system, water is processed in two ion exchange demineralizers with outputs of 7 gpm and 50 gpm.

In the water-to-water heat exchanger, the demineralized water is cooled from 140^{o} to 100^{o}F in a double pass through the U-shaped tubes. Cooling water circulates through the shell side of the exchanger. The unit is designed to transfer 18 x 10^6 Btu/hr with a cooling water temperature rise of about 25^{o}F.

The surge drum is a 7100-gal. vessel partially filled with demineralized water and pressurized with nitrogen to 150 psig. This arrangement prevents boiling in the gas-to-water heat exchanger in the event of a rapid heat load change and provides a safety margin in the event of pump failure (Figure III-9).

The energy in the demineralized water is transferred to the cooling water, as indicated above, in the water-to-water heat exchanger. The cooling water is circulated by either of two 2100-gpm pumps in a "closed" loop to an induced draft, single cell, redwood cooling tower. This unit is designed to reduce the temperature of the cooling water from about 95^{o}F to about 70^{o}F under the most adverse climatic conditions anticipated at NRTS. Cooling water make-up (to compensate for evaporation and windage losses in the cooling tower and normal system leakage) is provided from the utility water system.

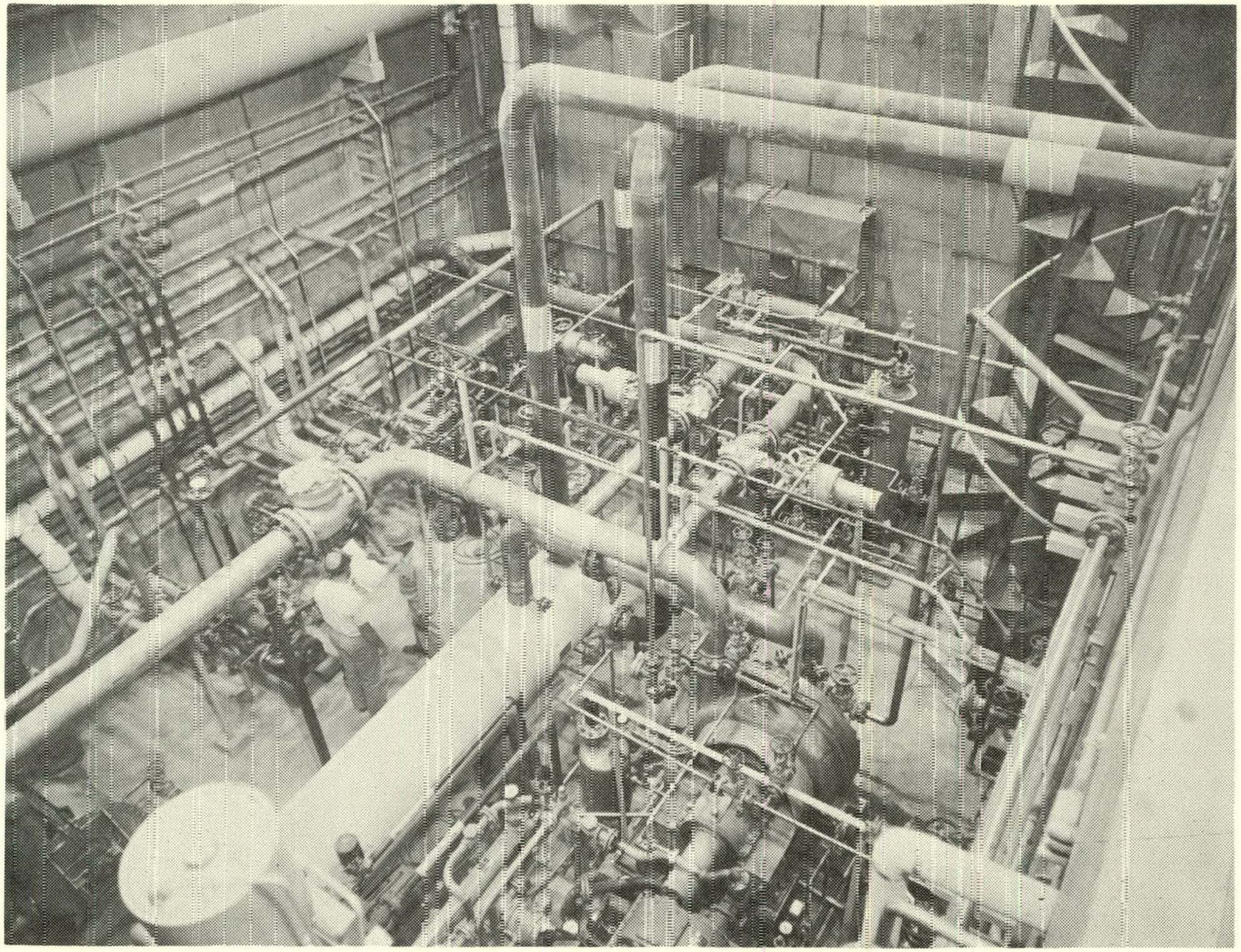

FIGURE III-5. GCRE MAIN COOLANT LOOP EQUIPMENT (LOOKING NORTH)

The gas-to-water heat exchanger is shown in the lower center; the gas filtering and drying equipment is in the upper center. The pool water and loop decontamination circulating pumps are shown on the left.

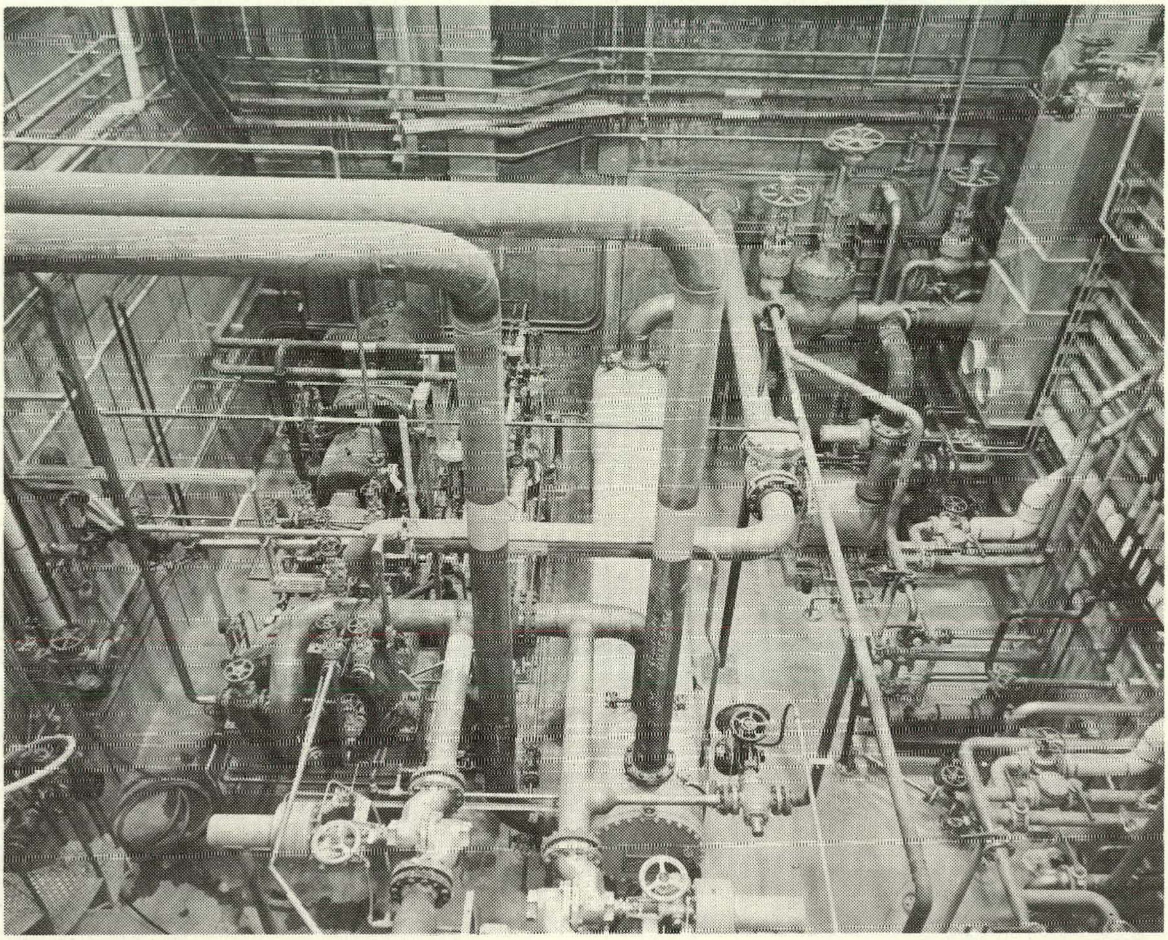

FIGURE III-6. GCRE MAIN COOLANT LOOP EQUIPMENT (LOOKING SOUTH)

The shutdown blower is on the right and the gas-to-water heat exchanger is shown in the center. The compressor is on the left; from foreground to background the units are the compressor, gear speed increaser, emergency drive motor, speed controller and drive motor.

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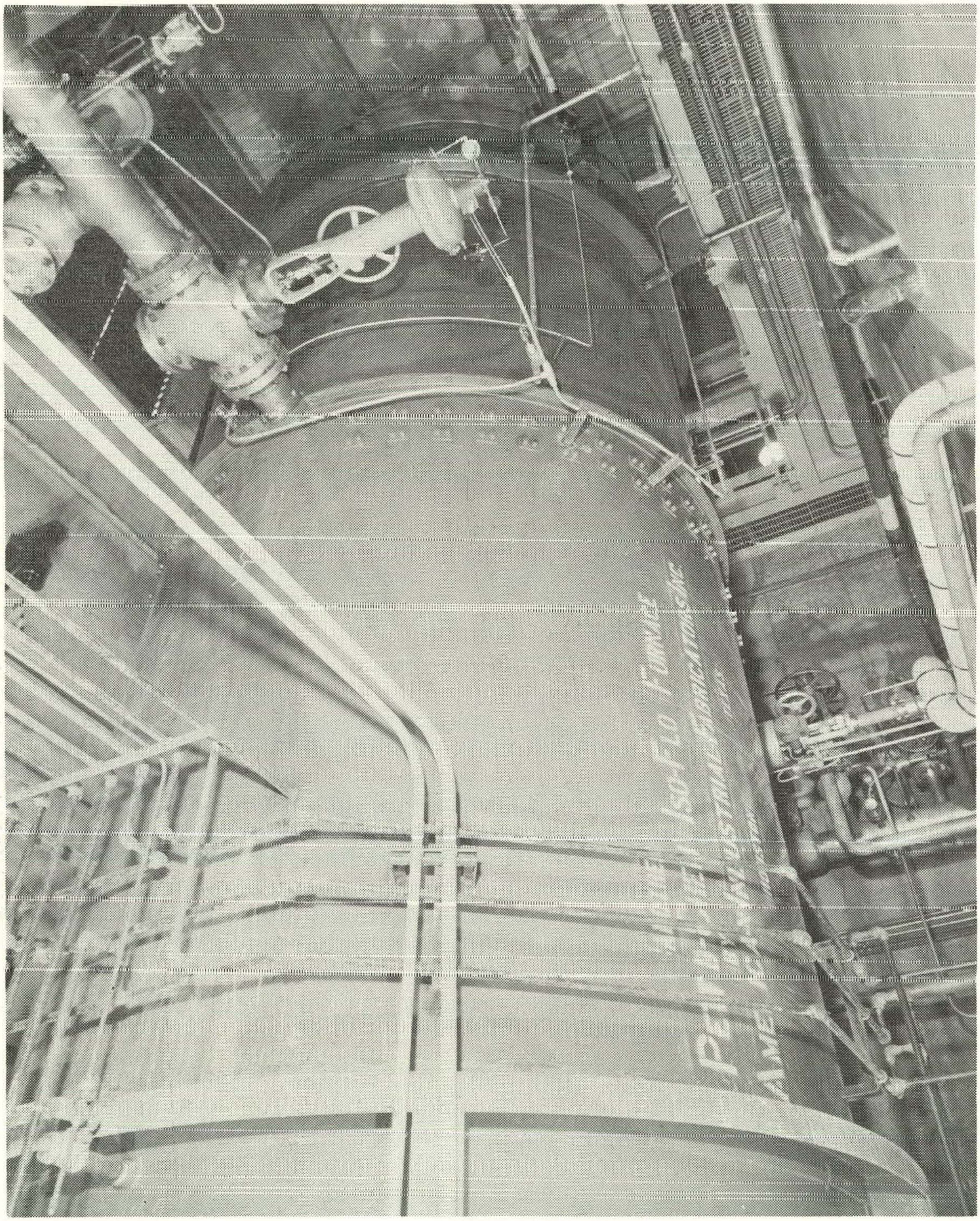

FIGURE III-7. OIL-FIRED GAS HEATER

View of the GCRE oil-fired gas heater from the floor of the heater pit. Bypass control valves are shown at the upper left.

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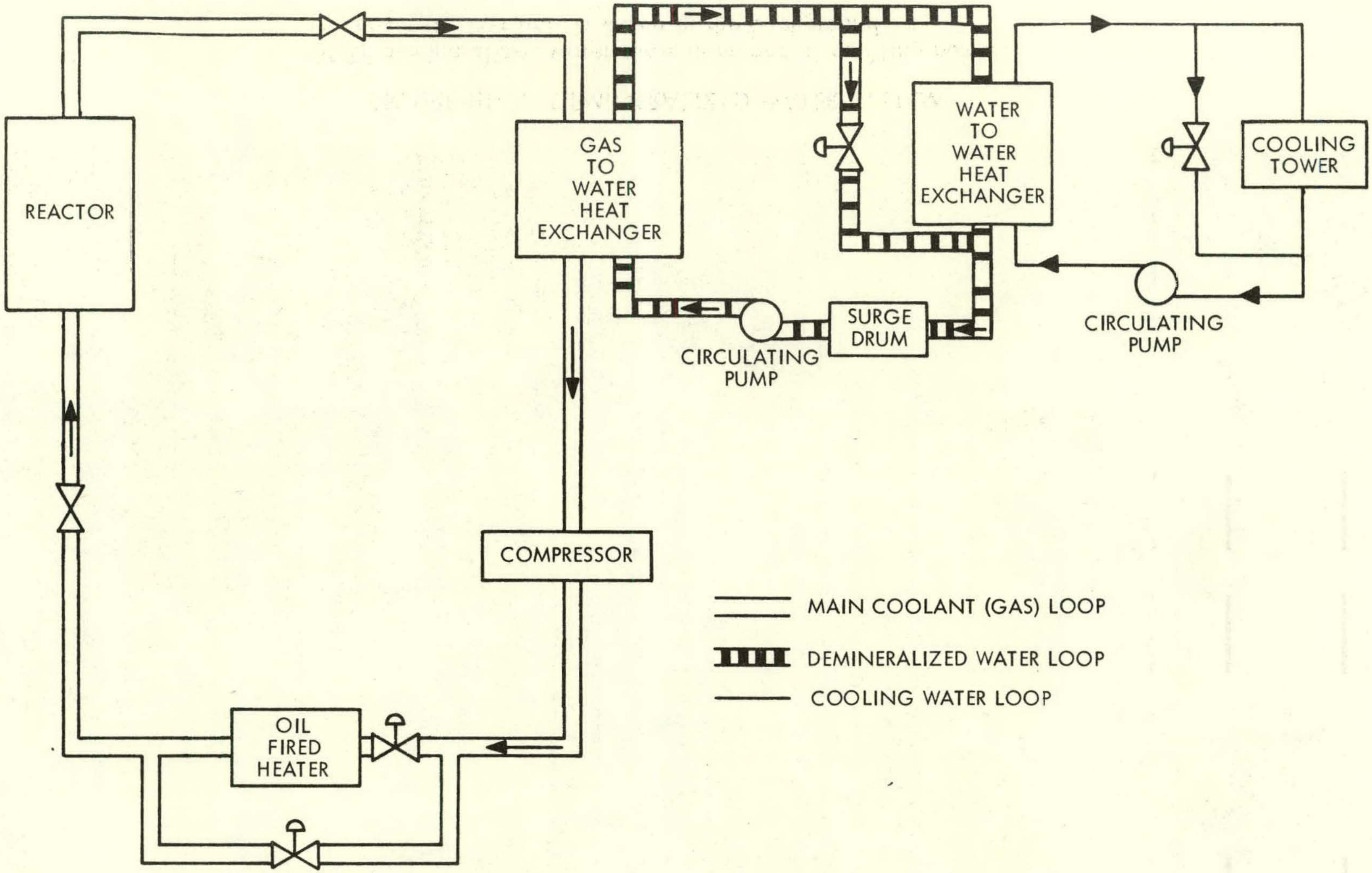

FIGURE III-8. GCRE HEAT REMOVAL SYSTEM, FLOW DIAGRAM

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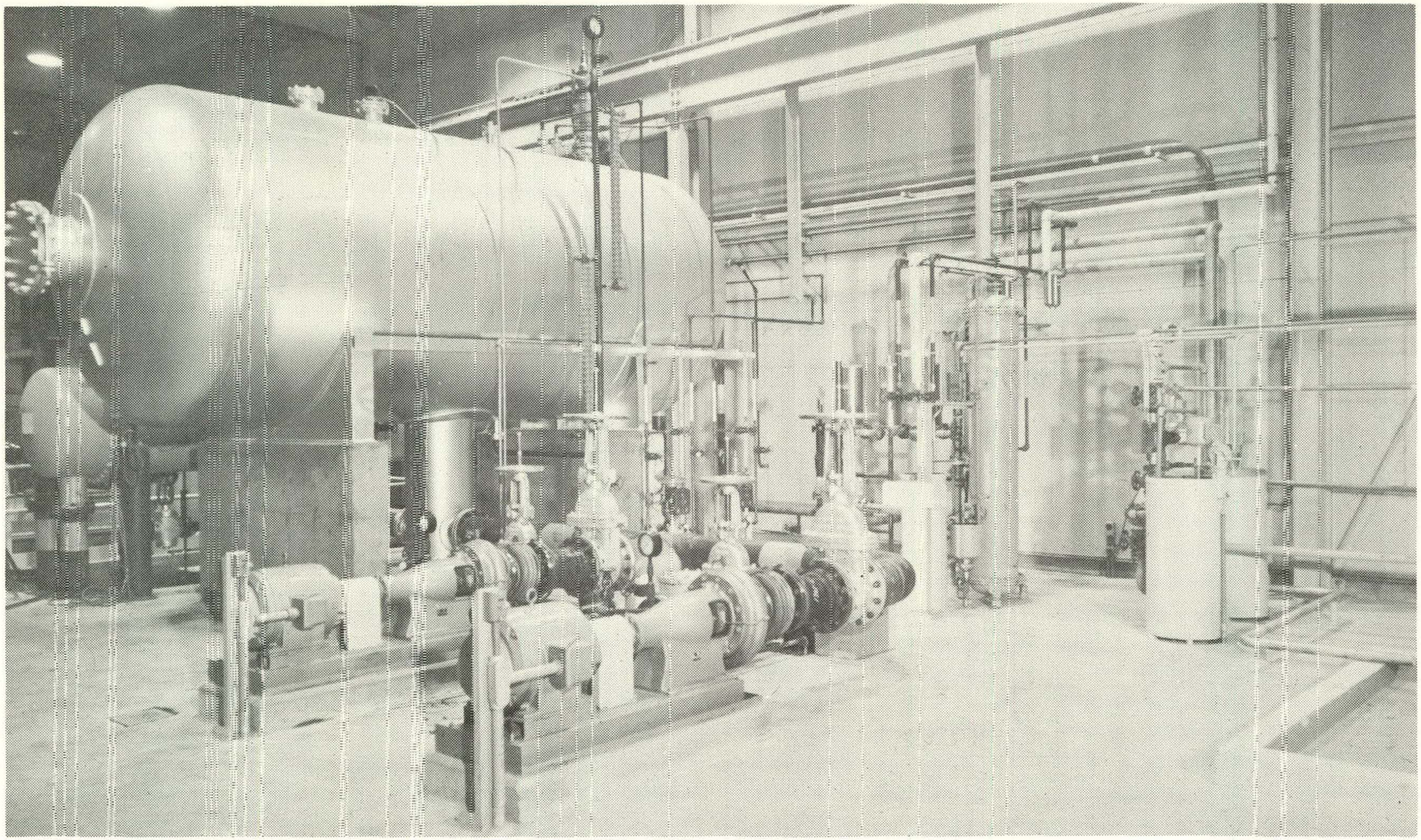

FIGURE III-9. DEMINERALIZED WATER SYSTEM

GCRE demineralized water surge drum and circulating pumps. The demineralizer unit is shown in the background.

Pressure and temperature instrumentation is provided throughout the demineralized water and cooling water loops to monitor the operation of these systems. All pertinent parameters are displayed in the control room.

3. Support Systems

The systems described below are provided to support the operation of the main coolant and heat removal loops.

a. Shutdown Cooling System

The shutdown cooling system is provided to circulate a small quantity of gas through the reactor to remove afterheat following a normal shutdown of the main coolant loop. The shutdown cooling system includes a gas blower and an air-to-air heat exchanger. The system is automatically actuated when the gas pressure in the main coolant loop is reduced to 10 psig. The blower circulates 380 cfm at a reference pressure of about 10 psig and the heat exchanger has a capacity of 300,000 Btu/hr when the inlet gas temperature is 1200^{o}F. The design discharge temperature from the heat exchanger is 350^{o}F.

This system is used primarily to control temperatures in the reactor when the block valves in the main coolant loop are closed and the external (heat exchanger-compressor-heater) portion of the loop is depressurized for maintenance or servicing.

b. Coolant Filtering and Drying System

Approximately 1% of the discharge flow from the main loop circulation compressor is diverted through equipment to filter and dry the gas. Filtration is provided in one of two units which have the capability of removing 0.3 μ or larger particles with an efficiency of 99.95%. Two dryer units are provided to reduce the dew-point of the 135-scfm gas flow to minus 60^{o}F. Each dryer is loaded with 130 lb of Linde molecular sieve adsorbent; each unit operates "on stream" for 8 hr and is regenerated in 8 hr. The filtered, dried gas is returned to the main coolant loop at the suction of the main compressor.

c. Pool Water System

The GCRE-I reactor operated submerged in a 20 ft by 20 ft by 30 ft deep pool filled with demineralized water. The water acted as the moderator for the reactor, as coolant for the calandria pressure tubes and tube sheets and as the primary radiological shield. The water was circulated by a 100-gpm pump and cooled in a 1,000,000-Btu/hr shell-and-tube heat exchanger. The design discharge temperature from the heat exchanger was 105^{o}F. The quality of the pool water was maintained by circulation through a 50-gpm mixed-bed demineralizer.

d. Instrument and Utility Air Systems

Filtered and dried instrument air is provided by a system including a compressor, filter, dryer, receiver and piping distribution system. The compressor is capable of delivering 40 cfm and the system controls regulate pressure between 75 and 90 psig.

The utility air system provides air primarily for atomizing the fuel oil for the oil-fired heater. A 365-cfm compressor, with interstage cooling, delivers air to the receiver. The control system maintains the system pressure between 115 and 125 psig.

Provision is made to manually connect the utility air system to the instrument air distribution piping in the event of the failure of the instrument air system. The main components of the systems are shown in Figure III-10.

e. Fuel Oil System

The fuel oil from the storage tank is pressurized to 115 psig by two 10-gpm pumps for distribution to the burners of the oil-fired nitrogen gas heater, the boiler and the oil-fired space heaters in the test building. A 100-gpm pump is provided to transfer oil from the transport tanker to the storage tank.

f. Electrical System

Normal power for the GCRE facility is provided by the 13.8-kv NRTS transmission system. A 1000-kva 13,800/480-v transformer is provided at the facility and the normal 480 v GCRE distribution network supplies all electrical loads.

Emergency power is provided by a 300-kw 480 v diesel-driven generator. This unit operated continuously when the reactor was operating and powered the critical instrument and process equipment loads to assure safe shutdown conditions in the event of a power failure. This unit is shown in Figure III-11.

g. Reactor Flooding System

Provision was made for the controlled flooding with demineralized water of the gas coolant passages and lower plenum of the reactor. Reactor flooding was required for fuel element handling and internal maintenance. The flooding system included a stainless steel tank, in which a large number of holes had been drilled at the level corresponding to the maximum safe volume of water, and associated piping and valves for admission of the water to the reactor. A small electric heater was provided in the line to heat the water entering the reactor to minimize the thermal shock to the fuel elements in the event that flooding was required prior to the complete cool-down of the core. A restrictive orifice was provided in the line to limit the rate of water admission (and, consequently, reactivity increase) to the core.

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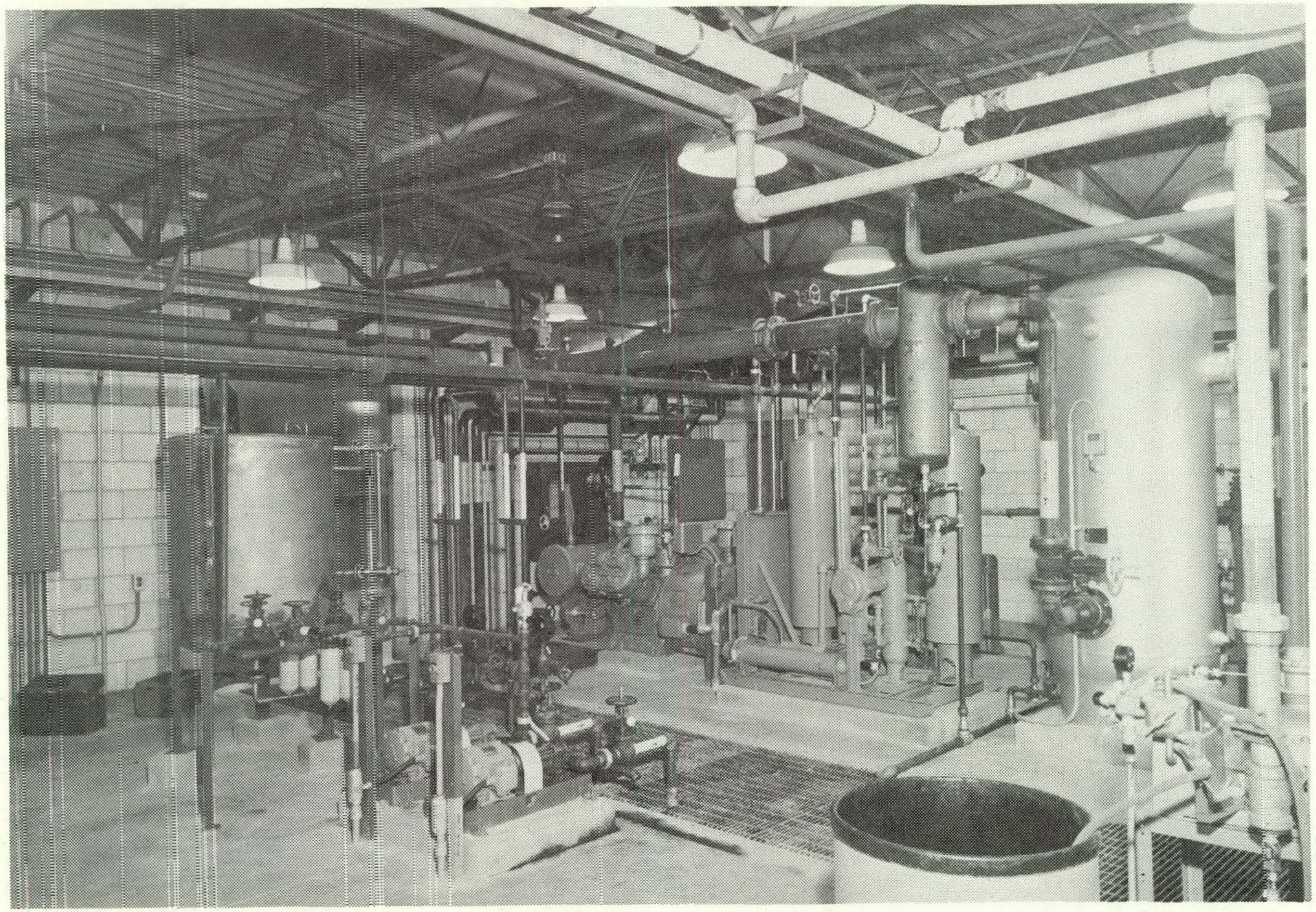

FIGURE III-10. GCRE MECHANICAL EQUIPMENT ROOM

The utility air compressor is shown in the left background; the instrument air compressor is partially obscured by the drying system to the right of the utility air compressor. The reactor flooding tank is on the left.

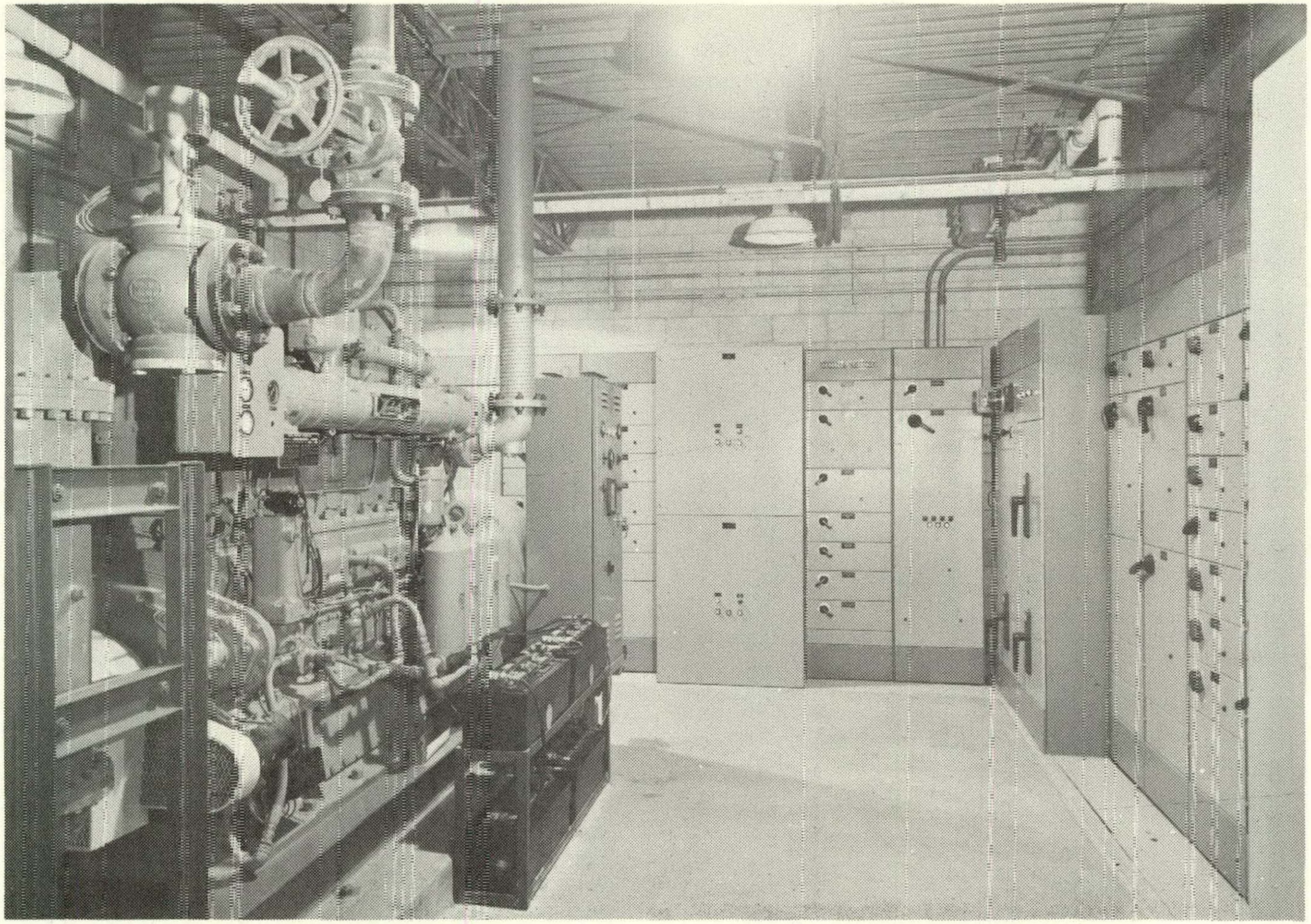

FIGURE III-11. GCRE ELECTRICAL EQUIPMENT ROOM

The diesel-generator unit is shown on the left; the generator is on the far end. Switchgear racks in the center and right contain motor starters, automatic transfer switches, etc.

h. Steam System

Provision was made to introduce steam from the facility boiler directly into the bottom of the reactor to act as a coolant in the event that the normal and emergency coolant circulation systems failed. This system was never operated during the GCRE program.

i. Chemical Decontamination System

Provision was made to circulate chemical decontamination solutions through the reactor loop and the external gas loop. Facilities were provided for mixing the solution and for forced circulation to the liquid waste tanks. This system was never used during the GCRE program.

IV. THE GCRE-I REACTOR

The GCRE-I reactor was a gas-cooled, water-moderated, fully enriched uranium-fueled reactor designed and fabricated to demonstrate the feasibility of the use of such a concept in a mobile nuclear power plant. The gas coolant entered the upper plenum chamber and flowed downward and was heated approximately 400°F in a single pass through the fuel elements. The fuel elements were supported in a calandria consisting of stainless steel tubes positioned at both ends by a stainless steel tube sheet. The reactor was submerged in and operated in a pool of demineralized water which served as the moderator and primary radiological shield. Control blades entered the reactor horizontally in the spaces between rows of calandria tubes. A lead fast neutron reflector surrounded the active portion of the reactor. The entire reactor assembly and control blade actuators were supported on a truss structure suspended from the walls of the pool. Provision was made in the upper plenum for the penetration of the pressure vessel by thermocouples from the instrumented fuel elements. The reactor assembly is shown in Figures IV-1 and IV-2.

The design operating level of the reactor was 2.2 Mw(t). The coolant gas entered at a nominal temperature of 800°F at 200 psia and left the reactor at a nominal temperature of 1200°F. The calandria provided positions for 73 fuel elements (or 72 elements and a neutron source). Twelve control rods were provided; nine of these were normally fully withdrawn during reactor operation. The normal coolant flow was 63,000 lb/hr of 99.5% nitrogen with 0.5% oxygen.

A. CALANDRIA

The original calandria for the GCRE-I reactor was fabricated from 6061-T6 aluminum forgings and tubes. The tube sheets were 3 in. thick and each tube was rolled into the sheet and welded at the outside surface to provide a gas-tight seal. Sixteen aluminum stay rods were installed near the circumference of the tube sheets to prevent overstressing of the pressure tubes. The pressure tubes were fabricated from 1.875-inch-OD seamless aluminum tubing with a wall thickness of 0.058 in. The tubes were arranged in a triangular lattice on 2.375 in. centers.

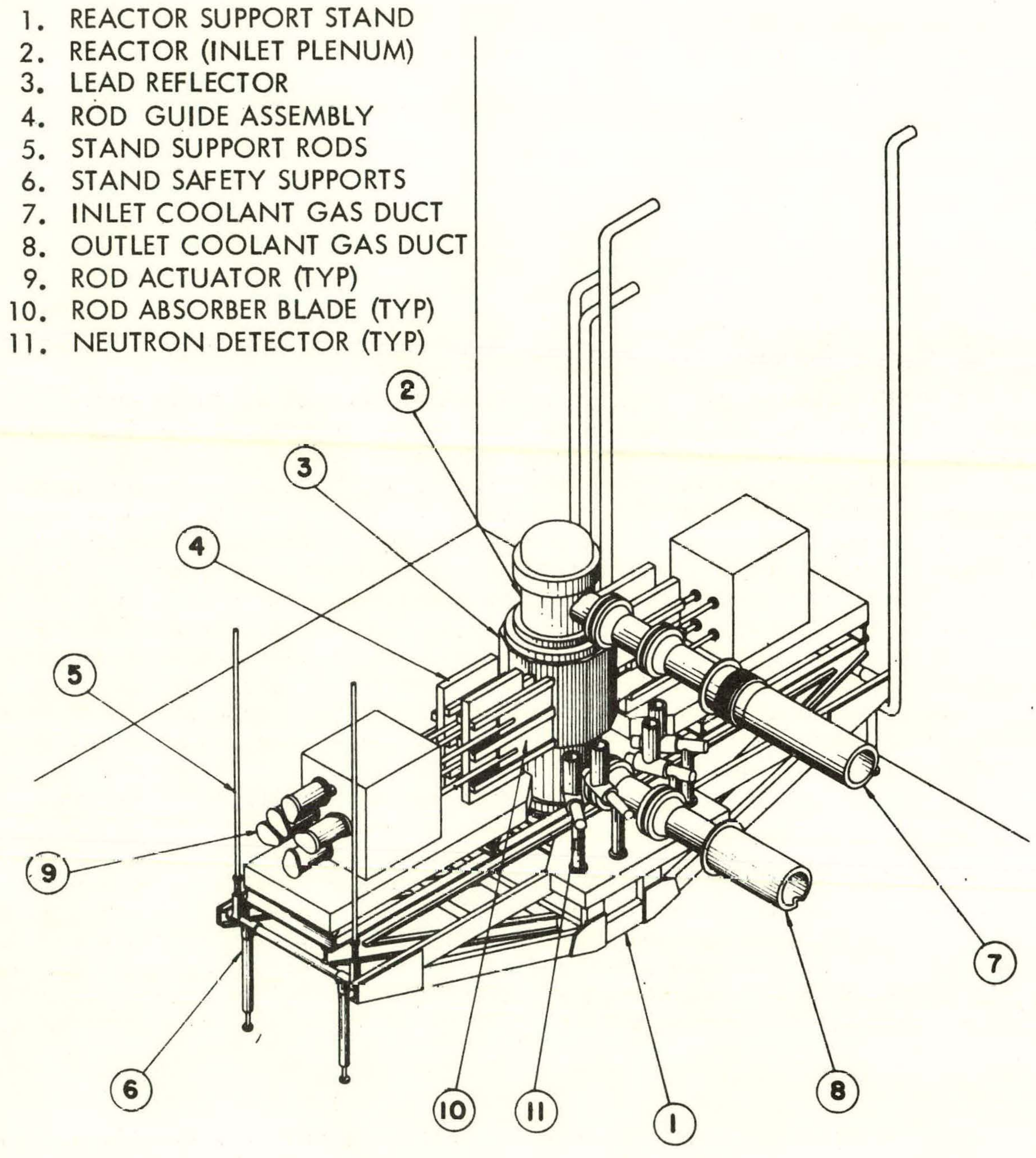

FIGURE IV-1. GCRE-I REACTOR ASSEMBLY

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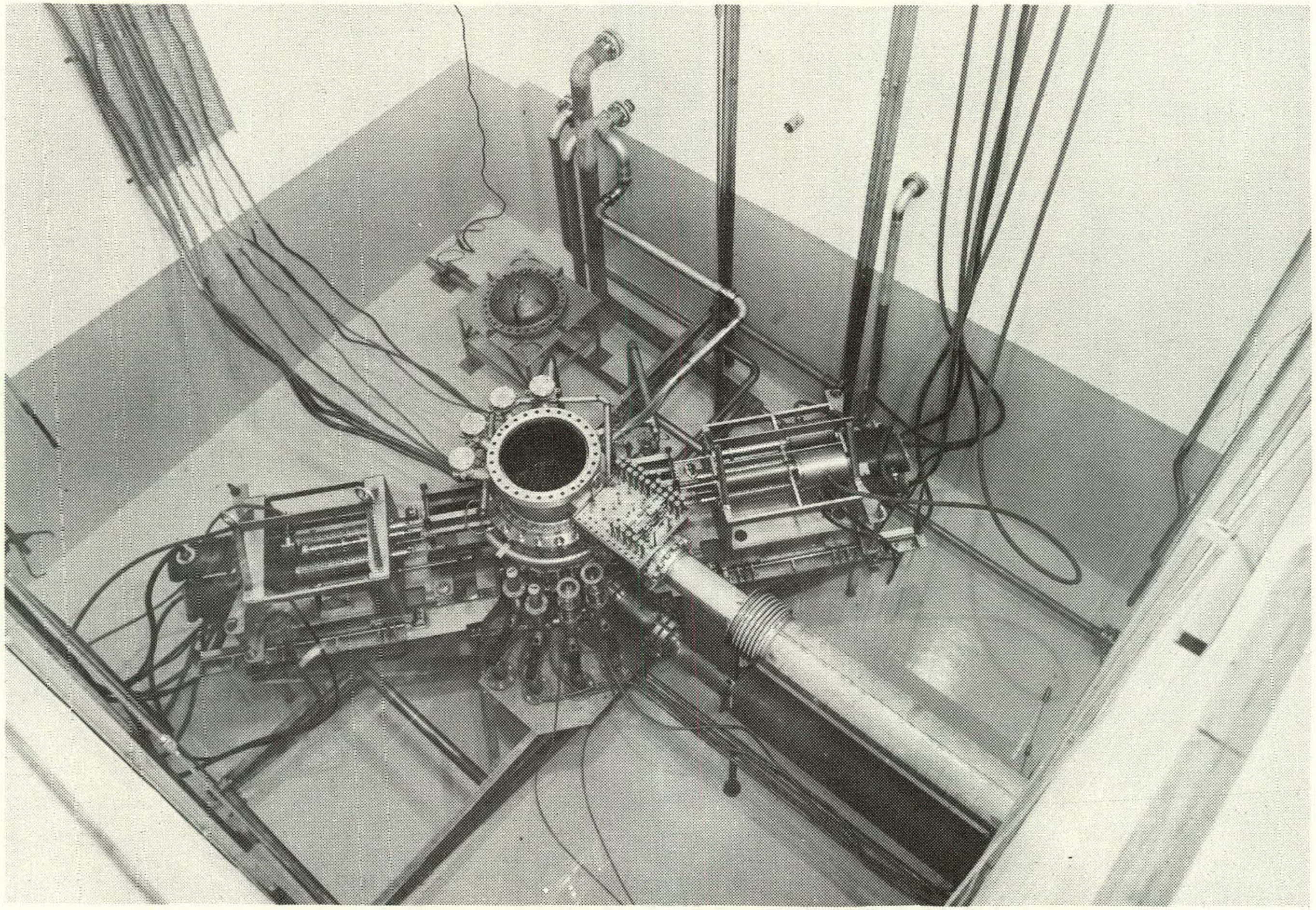

FIGURE IV-2. GCRE-I REACTOR ASSEMBLED IN REACTOR PIT

Note that the cover is removed and the pool partially filled with water. This view reflects the reorientation of the control rod assemblies which was made at the time of installation of the stainless steel calandria.

Although the calandria was subjected to thorough hydrostatic and helium leak checks at the completion of fabrication, the initial pressure checks following installation in the reactor approximately 10 months later revealed leakage at the seal welds on both ends of the unit. The calandria was removed from the reactor and a dye penetrant inspection revealed that a significant number of the seal welds were defective. Subsequent examinations indicated that the welds had failed because of an improper welding technique; a fusion weld on 6061 aluminum produces a brittle joint in which the natural crack at the weld root is propagated into the weld nugget. All of the original welds were machined off and the calandria was completely rewelded using a filler-type joint with 4043 welding rod.

Inspection following rewelding revealed several faulty welds which were subsequently repaired. Apparently, the repair work introduced stresses which resulted in the failure of welds adjacent to the reworked area. After considerable effort, the number of detectable leaks was reduced to one, at which time it was decided to reinstall the calandria for low-temperature, low-pressure nuclear testing and to proceed with the fabrication of a stainless steel tube bundle. This latter decision was made possible by the fact that the establishment of performance specifications for the ML-1 and the subsequent major change in the fuel element concept provided adequate reactivity to permit the use of stainless steel. A complete discussion of the aluminum calandria is given in Reference 10.

The design of the stainless steel calandria for the GCRE was based on the design of the calandria for the ML-1. The only significant differences were the diameter of the tube sheet, the number of tubes and the spacing between tube sheets. The GCRE stainless steel calandria consisted of seventy-three 1.760-in. ID pressure tubes fabricated from AISI Type 321 stainless steel tubing with a wall thickness of 0.020 in. The tube sheets were 3.5 in. thick and 40 in. in diameter and were fabricated from AISI Type 304 stainless steel. The distance between tube sheets was 30 in. Both tube sheets were rifle-drilled and counterbored (see Figure IV-3) prior to pressure tube insertion to provide water coolant passages to control the thermal stresses at acceptable levels. Water was circulated through these cooling passages by a submerged pump in the reactor pool which discharged through appropriate stainless steel piping to the inlet of each coolant passage. The effluent water from the coolant passages discharged directly to the reactor pool.

B. PRESSURE VESSEL AND REACTOR ASSEMBLY

A stainless steel plenum was bolted to the calandria at each end to complete, with the reactor cover, the pressure vessel assembly, as shown in Figure IV-4. (Plenums for both pressure vessels were stainless steel.) Both plenums and the inlet and outlet gas ducts were lined with laminated stainless steel radiation barrier type insulation (see Figure IV-5) to minimize heat transfer from the hot coolant gas through the plenum and duct walls to the pool water. The bolted flanges in the assembly (cover-to-upper plenum, upper plenum-to-calandria, calandria-to-instrumentation flange and instrumentation flange-to-lower plenum) were sealed with Type 3003 aluminum gaskets. The mating flange faces were machined to form serrated tongue and groove closures. The gas duct flanges were sealed with bolted stainless steel RTJ

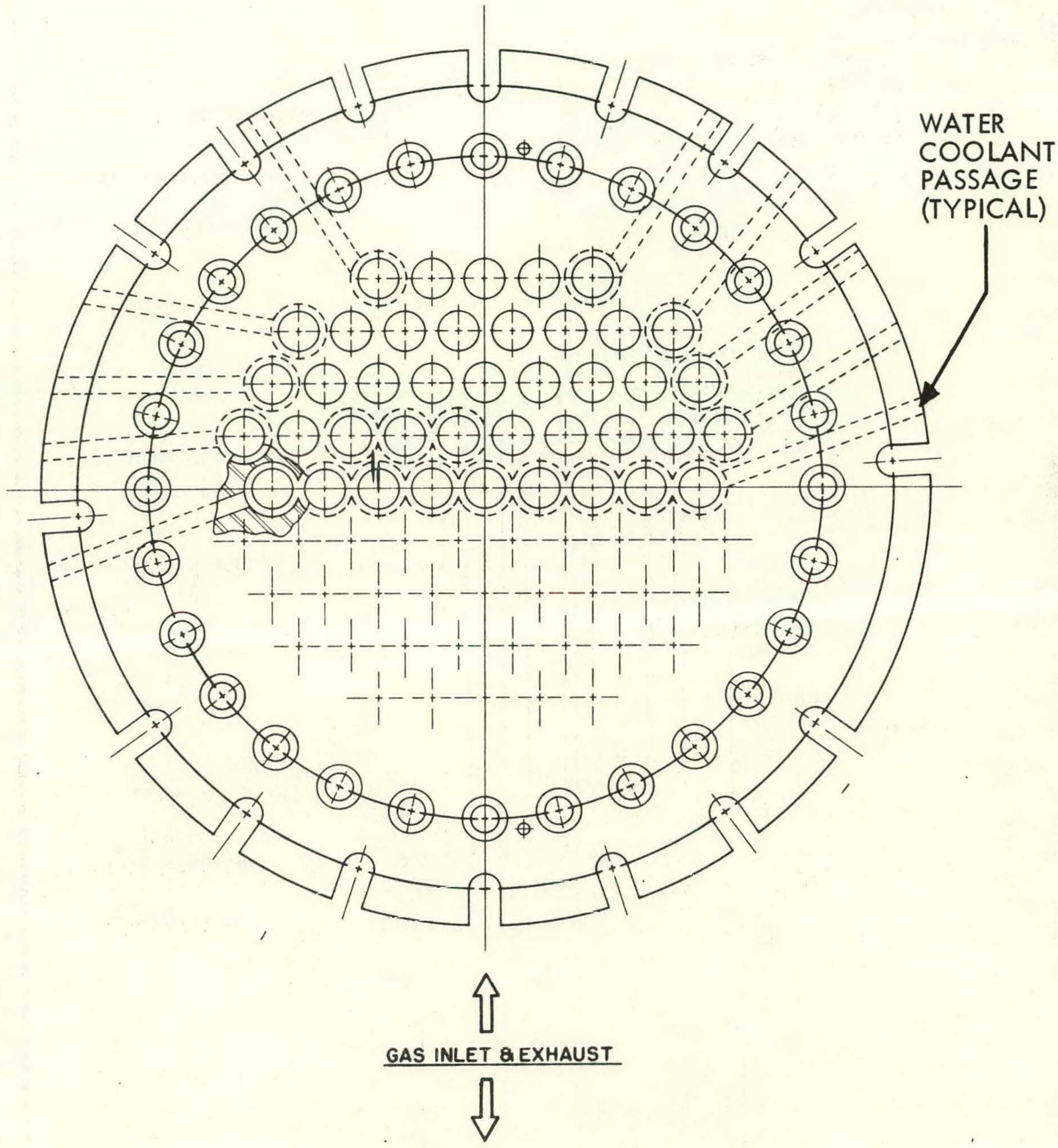

FIGURE IV-3. GCRE CALANDRIA (PLAN VIEW)

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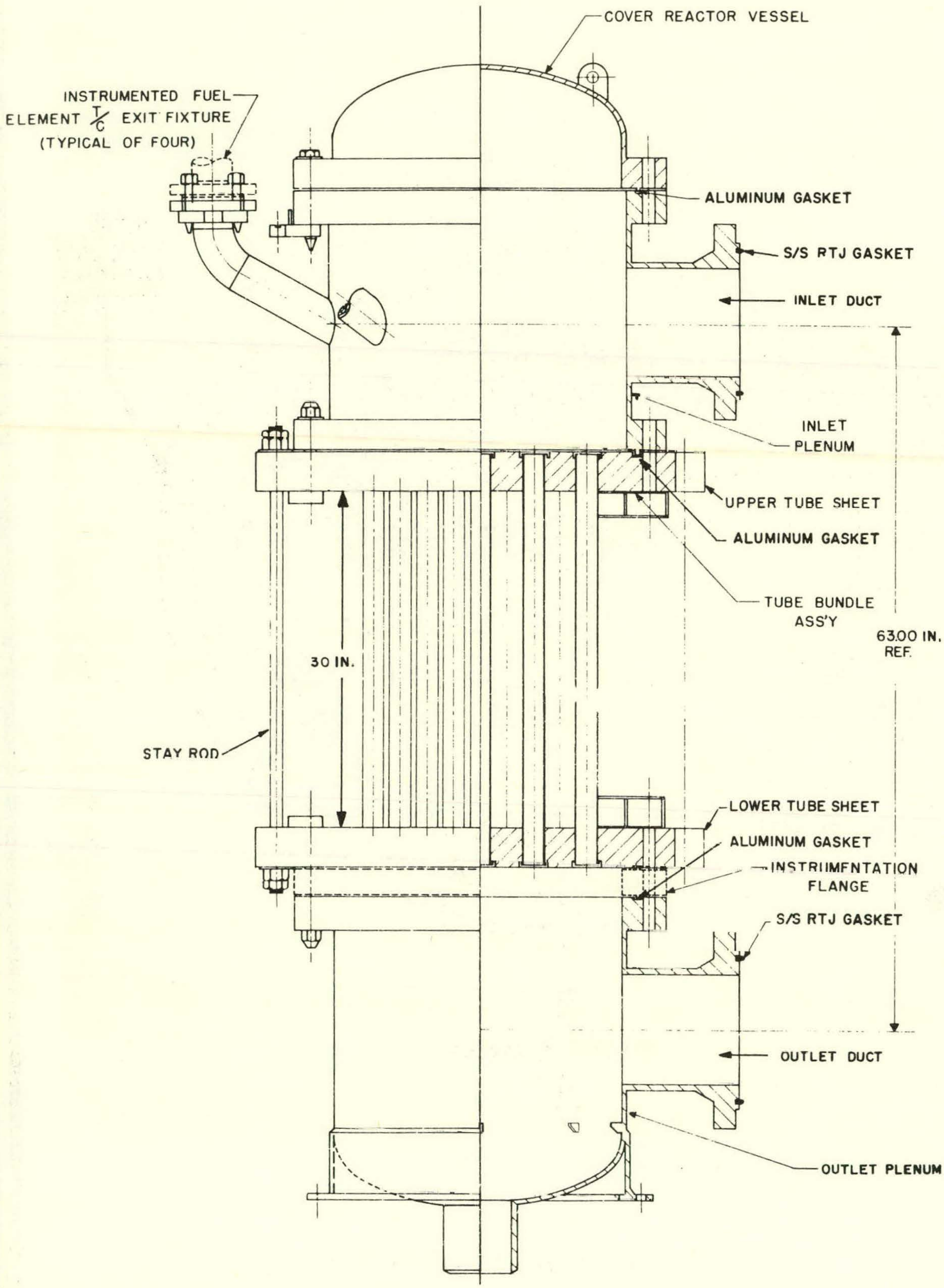

FIGURE IV-4. GCRE-I REACTOR PRESSURE VESSEL

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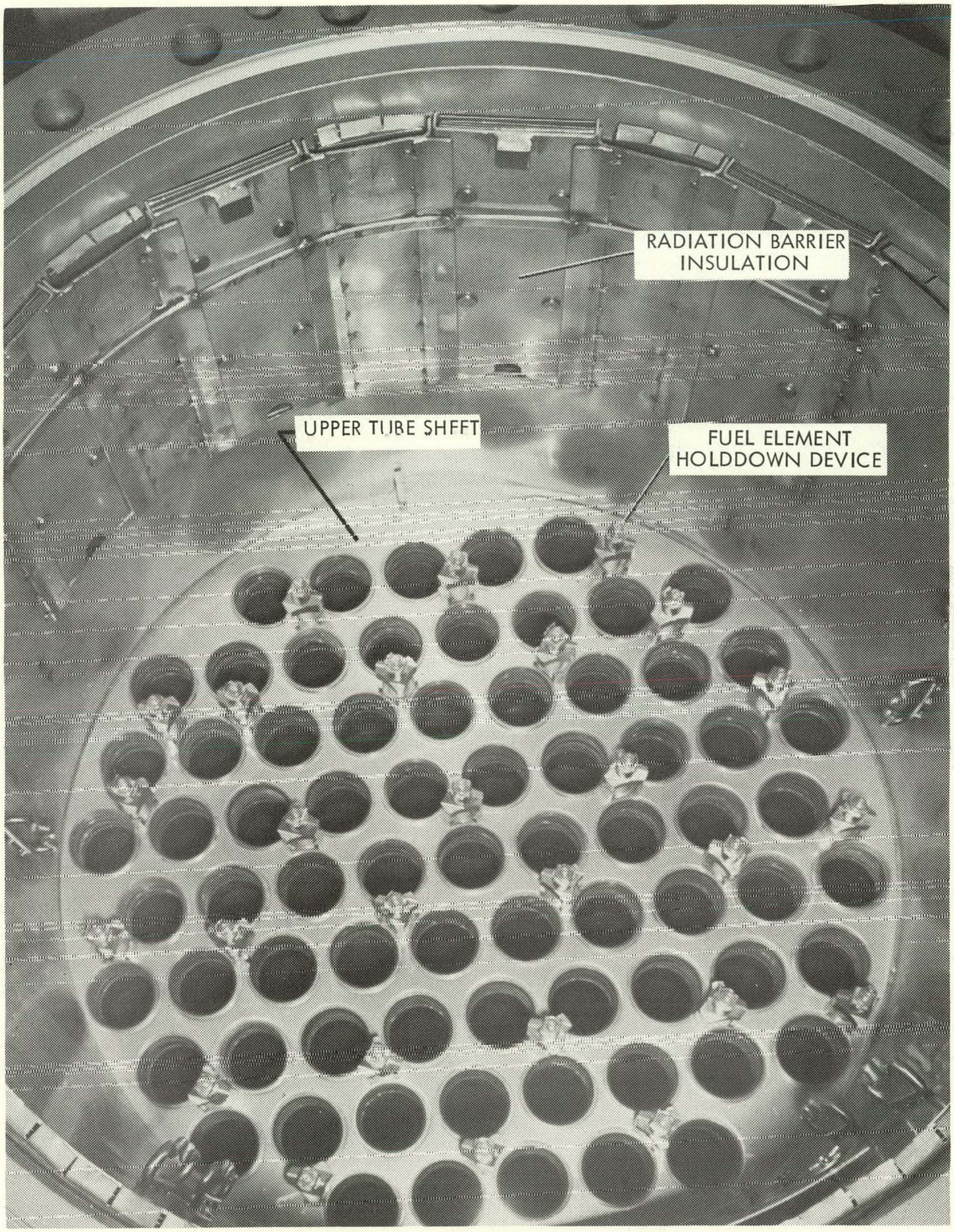

FIGURE IV-5. GCRE-I UPPER PLENUM

Note the radiation barrier insulation around the periphery of the plenum. The fuel element holddown devices are shown mounted on the upper tube sheet.

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closures. Piping was provided in the lower plenum for flooding and draining the gas passages (for fuel element handling or internal maintenance) and for the introduction of steam as an emergency coolant.

An instrumentation flange at the lower end of the calandria positioned thermocouples at the discharge end of each fuel element position and flow measuring (pitot) tubes under selected fuel element positions. The temperatures and pressures detected by these devices were indicated in the control room. In addition, the instrumentation flange included sampling tubes which permitted monitoring of the effluent gas from each fuel element for fission product activity (to locate a failed fuel element, if necessary).

The pressure tube section of the calandria (the active portion of the core) was surrounded with a 4-in. thick fast neutron reflector and primary gamma shield. This assembly was fabricated from high purity lead and was coated to prevent corrosion. The reflector consisted of several sections to facilitate fabrication and installation. Each section included several vertical tubes through which cooling water was pumped (from the pool circulation system) to control the temperature of the lead. The reflector was penetrated at appropriate places for the entry of the control blades.

The moderator water in the active section of the reactor core circulated primarily by convection. Cool water entered the region through a gap between the bottom of the reflector and the upper face of the lower tube sheet and flowed upward through the pressure tube region of the calandria. The warm water emerged from the core section through the gap between the top of the reflector and the lower face of the upper tube sheet where it was collected in a ring manifold and routed through the pool water heat exchanger to the suction of the pool water circulating pump.

C. CONTROL RODS

The GCRE-I control rod system consisted of 12 rods arranged in two banks of six each on opposite sides of the reactor (Figure IV-1). A summary of the designations, functions, locations and characteristics of the rods is presented in Table IV-1. The rod banks were mounted on the reactor support stand but were removable with remote tooling for maintenance and modification.

The design of the control rod system was based on several specific neutronic characteristics of the GCRE-I reactor. These are discussed below.

1) Since the fuel elements were thermally insulated from the moderator water, very little energy from the fuel would be deposited in the moderator and, for this reason, a nuclear excursion would not be self-limiting short of gross deformation of the core. For this reason, it was required that the control rod system design provide for fast shutdown under any credible set of circumstances (safety rods).

2) The GCRE-I reactor was undermoderated and it was predicted that an increase in reactivity of approximately 6.5% $\Delta K/K$ would be associated with water flooding of the gas passages. It was required that the control system be capable of controlling this reactivity increase (shutdown rods).

TABLE IV-1

GCRE-I CONTROL RODS

Designation	Function	Location		Position	Scram Time, Sec.	Stroke, in.	With-drawal Time, Min.	Unshadowed Worth, % $\Delta K/K$	Absorber
		Rack	Elevation						
X-1	Safety Rod	West*	Top	Center	0.170	15	12.5	1.5	Cd, Mallory***
X-2	Safety Rod	East**	Bottom	Center	0.170	15	12.5	1.5	Cd, Mallory
SD-1	Shutdown Rod	W	Top	Left****	0.500	15	12.5	1.5	Cd, In, Mallory
SD-2	Shutdown Rod	W	Bottom	Center	0.500	15	12.5	1.5	Cd, In, Mallory
SD-3	Shutdown Rod	W	Top	Right	0.500	15	12.5	1.5	Cd, In. Mallory
SD-4	Shutdown Rod	E	Bottom	Left	0.500	15	12.5	1.5	Cd, In, Mallory
SD-5	Shutdown Rod	E	Top	Center	0.500	15	12.5	1.5	Cd, In, Mallory
SD-6	Shutdown Rod	E	Bottom	Right	0.500	15	12.5	1.5	Cd, In, Mallory
S-1	Shim Rod	W	Bottom	Left	--	18	23.5	1.0	Mallory
S-2	Shim Rod	W	Bottom	Right	--	18	23.5	1.0	Mallory
SB-1	Setback Rod	E	Top	Left	0.3	18	1.5	0.3	Mallory
F-1	Fine (Regulating) Rod	E	Top	Right	--	18	0.7	0.4	Mallory

* Later reoriented to NW

** Later reoriented to SE

*** Mallory is a tungsten-copper-nickle alloy approximately 90-95% Tungsten

****As viewed looking toward the reactor from behind either rod bank

3) The contemplated mode of operation for the GCRE-I involved fuel element temperatures which very closely approached the maximum allowable temperature. This condition required that an automatic means of reducing reactor power be provided. In the interest of maximizing reactor "on-stream" time, it was desirable that this reduction not result in an immediate reactor shutdown and that resumption of power operation be possible under a straightforward procedure (setback rod).

4) The rods normally in the reactor during operation should affect the flux and temperature distribution as little as possible. (This affected the design and location of the shim and fine rods.)

5) The control system should be as flexible as possible to permit the maximum utilization of the GCRE-I as an experimental and developmental tool. (This resulted in the use of many (12) rods as opposed to a smaller number of more absorbent rods.)

The following mechanical characteristics were common to all rods:

1) Each rod had an absorber blade which was mounted vertically and traveled horizontally in guides in the spaces between rows of calandria pressure tubes. The size and composition of the blades varied with the function of the rod and the size and material of the guide varied with the size and weight of the blade.

2) All absorber blades were connected to the actuator shaft with a latch assembly which could be disengaged remotely.

3) All actuator mechanisms were sealed in internally pressurized water-tight housings. Double sliding contact seals were provided at the point where the actuator shaft penetrated the housing.

4) Microswitches were provided in the actuator assembly to de-energize the drive motor and actuate an indicating circuit at each extreme of rod travel. These microswitches also functioned in the rod withdrawal sequence interlock circuit.

5) The electrical and control wiring for the actuator was contained in a flexible, armored, waterproof sheath sealed to the actuator housing and suspended from the pool wall above the water level (see Figure IV-2).

6) The normal (as opposed to scram) motion of all rods was accomplished by an electric motor through a lead screw and nut. A typical control rod actuator is shown in Figures IV-6 and IV-7.

In the safety rods, the blade and actuator shaft were coupled to the lead screw drive assembly with an electromagnetic ball latch (see Figure IV-8). When energized, this latch connected the drive and blade assemblies securely so that normal withdrawal or insertion of the blade was possible with the motor drive. In the event of a scram condition, the ball latch was de-energized, permitting the internal pressure in the housing to act on the actuator shaft assembly and drive the blade into the reactor. Dashing at the end of the stroke was accomplished by pool water acting in a perforated cylinder arrangement.

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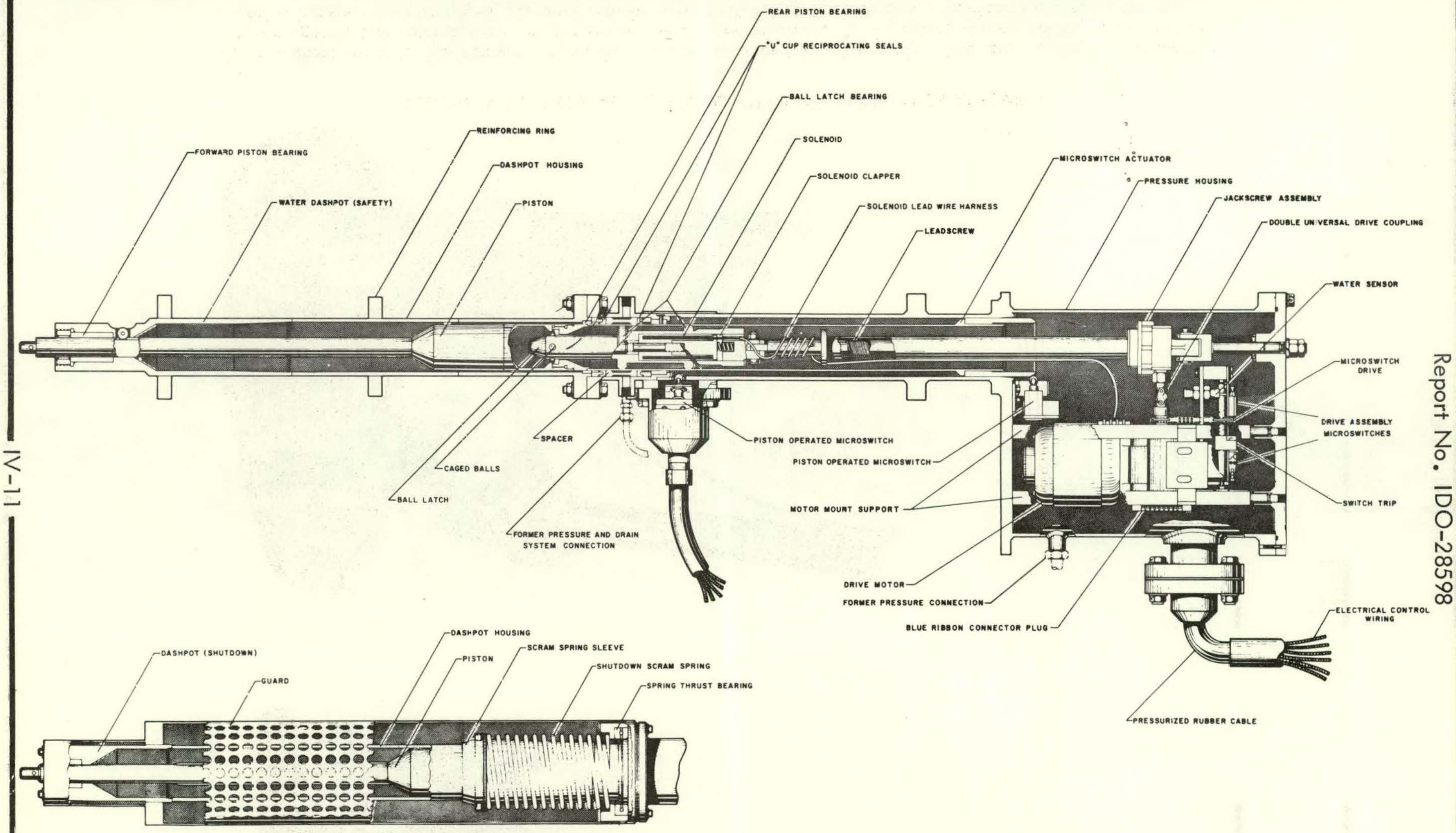

SHUTDOWN MECHANISM

SCRAM SPRING ARRANGEMENT

FIGURE IV-6. GCRE-I SAFETY AND SHUTDOWN ROD ACTUATOR

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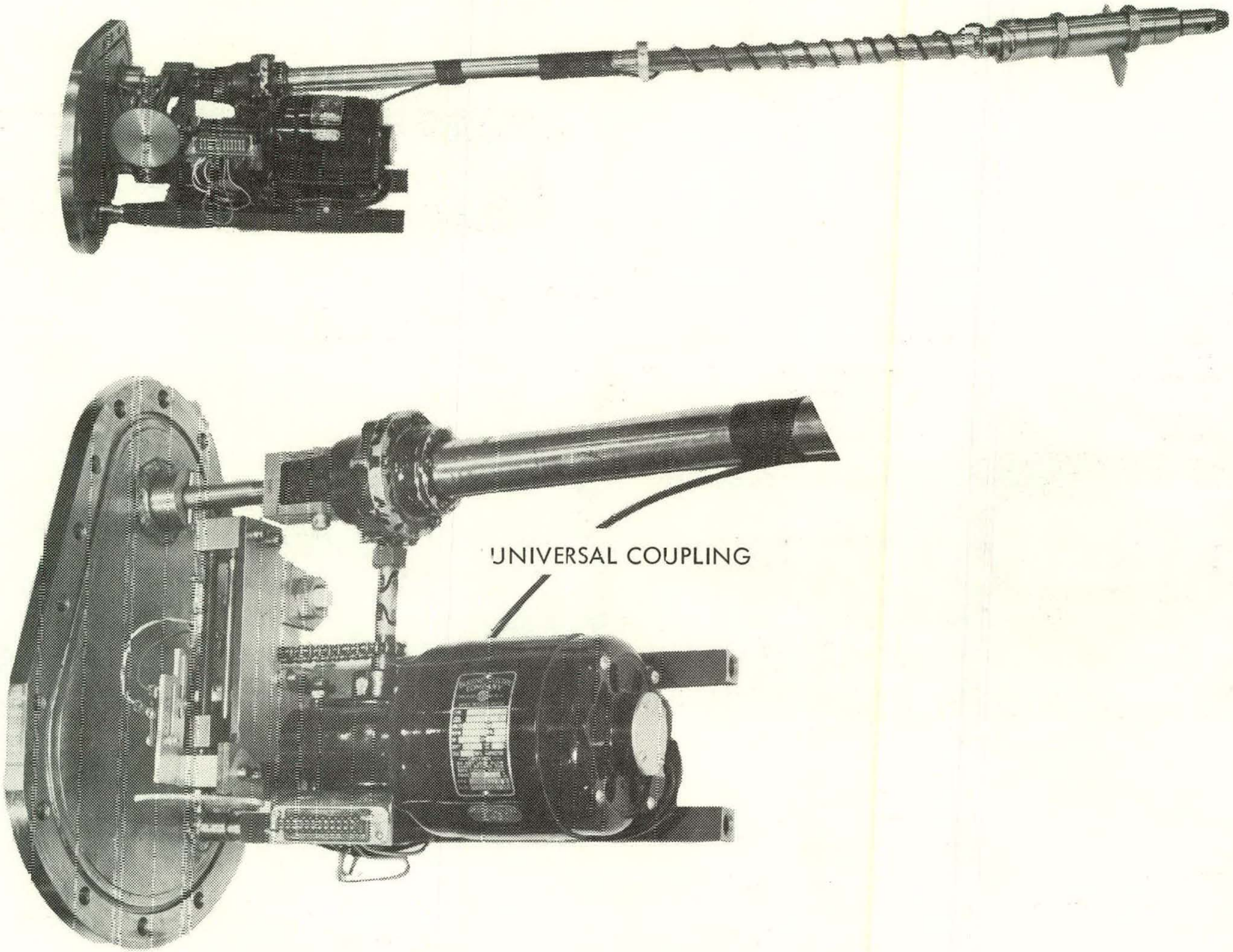

FIGURE IV-7. TYPICAL GCRE-I CONTROL ROD ACTUATOR ASSEMBLY

This is a shutdown rod. The top view shows the actuator motor and drive assembly. The scram spring is distended; the electromagnetic ball latch is shown at the extreme end of the actuator. In the enlarged view below, the universal coupling between the motor/gear assembly and the unit which drives the lead screw (enclosed) is clearly shown.

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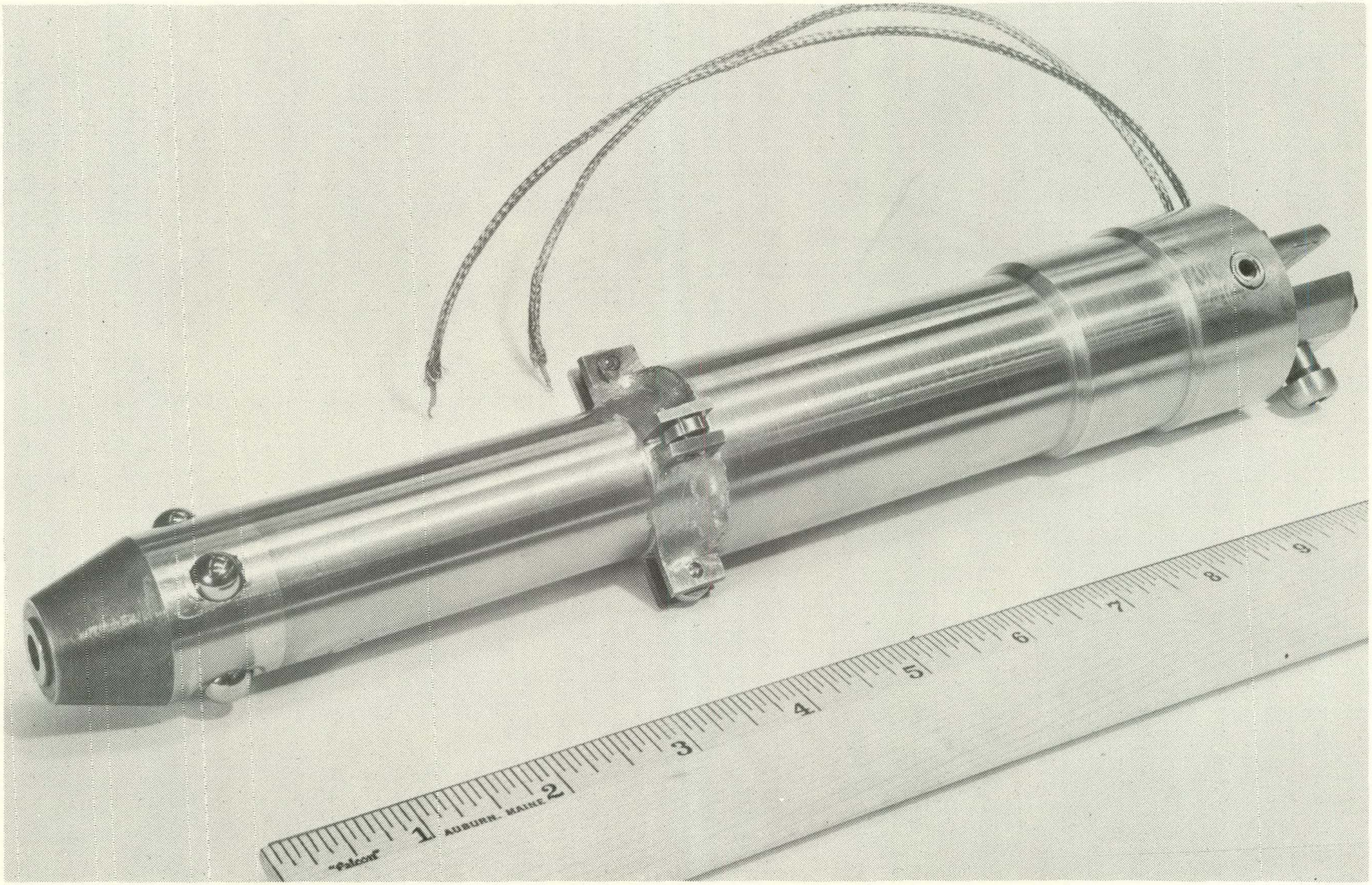

FIGURE IV-8. GCRE-I CONTROL ROD ELECTROMAGNETIC BALL LATCH

In the shutdown rods, the scram action was provided by a compressed spring; the ball latch arrangement was the same as in the safety rod. The setback rod was "scrammed" by a high speed, non-reversing motor operating through a magnetic clutch.

D. INSTRUMENTATION

The design of the GCRE Test Building permitted the safe entry and occupancy of the reactor and equipment areas during normal operation of the reactor. However, the possibility of a radiation-producing malfunction precluded the permanent assignment of personnel to work stations in the test building. As a consequence, provision was made in the control room to monitor all reactor and process operating parameters and to operate all essential process equipment. This design philosophy resulted in a relatively large array of instrumentation and controls as shown in Figure IV-9.

It is not considered appropriate to describe the function of each instrument and control circuit in this report; such information can be inferred from the legend of Figure IV-9 and Appendix D. The major instrumentation systems are described below:

1. Nuclear Instrumentation

Nine channels of nuclear instrumentation were installed to monitor and control the operation of the GCRE-I reactor. Detectors were located around the reactor (as shown in Figure IV-1) and high voltage and signal cables were routed out of the reactor pool in metal-sheathed, pressurized cables. The description and function of the nuclear instrumentation channels is presented in Table IV-2. This entire system was purchased as a package from the Minneapolis-Honeywell Co. and duplicated, with some improvements, systems installed by this vendor on other reactors (MTR, for example). The channel outputs were displayed on meters and recorders in the control room.

2. Gas Temperature Monitor

The thermocouples at the effluent end of each fuel element (see earlier discussion of instrumentation flange Section IV-B) were cabled to a patch panel in the control room. From this panel, selected thermocouple outputs could be displayed on a 75-point recorder, an 18-channel recording high-speed oscillograph, a 98-channel oscilloscope (to present a profile of selected temperatures) or a single point indicator with an annunciator (normally used for highest observed temperature).

3. Loop Controls

The significant temperatures, flows and pressures in the main coolant loop were displayed in the control room and provision was made for the control of these parameters with appropriate circuitry.

4. Control Rods

All control rods were operable from the main console. In addition, the nitrogen (scram) pressure in the safety rod housings and the sensors which sensed water in the rod housings were monitored by control room instrumentation.

TABLE IV-2

GCRE-I NUCLEAR INSTRUMENTATION

Channel Designation	Function	Range, watts	Detector	Amplifier	Scram Circuits	Interlock Circuits
I	Log Count Rate/ Period (Start-up Range)	0.001 to 50	Fission Chamber	Linear pre-amplifier & amplifier with period detection	Setback on period shorter than 10 sec	Permissive startup interlock on detectable count rate
II	<---SAME AS CHANNEL I--->					
III	Log-N/period (Low End of Operating Range)	(6.3 decades)*	Compensated Ion Chamber	Log-N/period amplifiers	Scram on period shorter than 7 sec; scram at 115% power	Disconnects channel I & II high voltage at 5% power
IV	<---SAME AS CHANNEL III--->					
V	High level safety (High End of Operating Range)	5×10^4 to 10^7	Parallel Circular Plate Ion Chamber	Composite safety amplifier	Setback at 110% power; scram at 115% power	None
VI VII VIII	<---SAME AS CHANNEL V--->					
IX	Linear power system (auto control)	5 to 10^7	Compensated Ion Chamber	Electrometer	None	None

*Positioned to cover full operating range after startup.

LEGEND FOR FIGURE IV-9 - GCRE CONTROL ROOM LAYOUT

RACK	PANEL	INSTRUMENT(S) OR CONTROL(S) FUNCTION
1	---	Control Relays
2	---	Control Relays
3	A	Wind Speed and Direction Indicators
3	B	Main Compressor Controls
3	C	Cooling Water and Lube Oil Pump Controls
3	D	Main Compressor Controls
3	E	Heater Controls
3	F	Remote Operable Valve Controls
4	A	Shutdown Cooling Loop Temperature Indicator
4	B	Shutdown Cooling Loop Flow Indicator
4	C	Fire Pump and Test Building Exhaust Fan Controls
4	D	Well, Sump and Wash Pump Controls
4	E	Stack Blower, Pool Water Pump and Wash Pump Controls
4	F	Reactor Flooding Water Level Indicator
4	G	Water-in-Rod Housing Indicators
5	A	Annunciator Panel
5	B	Water-to-Water Heat Exchanger Bypass Control (Temperature)
5	C	Cooling Tower Bypass Control (Temperature)
5	D	Pool Water Heat Exchanger Bypass Control (Temperature)
5	E	Heater Bypass (Reactor Inlet Temperature) Control
5	F	Make-up Gas Flow Indicator
6	A	Annunciator Panel
6	B	Main Loop Gas Dew-Point Indicator
6	C	Main Loop Oxygen Concentration Indicator
6	D	Filter/Dryer Loop Flow Indicator
6	E	Waste Gas Holder Level Indicator
7	A	Log Count Rate Recorder
7	B	Scaler
7	C	Log Count Rate (Channel I) System
7	D	Log Count Rate (Channel II) System
7	E & F	Linear Amplifier
8	A	Period Recorder
8	B	Log-N Period (Channel III) System
8	C	Log-N Period (Channel IV) System
8	D & F	Period Amplifier (Channels III and IV)
8	F & G	Compensated Ion Chamber Power Supply
9	A	Log N Recorder
9	B	Linear Power Recorder (Channel IX)
9	C	Linear Amplifier (Channel IX)
9	D	Micro-micro Ammeter (Channel IX)
9	E	Compensated Ion Chamber Power Supply
10	A	Power Level Recorder
10	B	Safety Channel Amplifier and Trip Circuit (Channels V and VI)
10	C	Safety Channel Amplifier and Trip Circuit (Channels VII and VIII)
10	D & E	Scram Timer
10	F	Safety Rod Housing Pressure Indicator
11	A	Annunciator Panel
11	B	Main Loop Flow Indicator
11	C	Reactor Outlet Temperature Indicator
12	A	Annunciator Panel
12	B	Reactor ΔP Indicator
12	C	Reactor Inlet Pressure Indicator
13	A	Annunciator Panel
13	B	Reactor ΔT Indicator
13	C	Heater Outlet Temperature Indicator
13	D	Main Compressor Outlet Temperature Indicator
13	E	Compressor Inlet and Outlet Pressure Indicator
14	A	Gas-to-Water Heat Exchanger Outlet Gas Temperature Indicator
14	B	Gas-to-Water Heat Exchanger Outlet Gas Temperature Controller
14	C	Compressor Seal Bleed Flow Controller
14	D	Compressor Seal Pressure Indicator
15	A	Btu Recorder
15	B	Pitot-Static Flow Meter Controls
15	C	Gas-to-Water Heat Exchanger Inlet Water Temperature Indicator
15	D	Gas-to-Water Heat Exchanger Outlet Water Temperature Indicator
15	E	Demineralized Water Flow Indicator
15	F & G	Power Supplies
16	A	Fission Product Sampler Count Rate Indicator
16	B & C	Fission Product Monitor System
16	D	Loop Purge Flow Control
16	E	Power Supply
17	A, B, & C	Fuel Element Temperature Indicators
17	D, E, & F	Remote Area Radiation Monitors
18	A & B	Fuel Element Temperature Indicators
18	C	Recording Oscillograph
18	D & E	Miscellaneous Temperature Meters and Selector Switches
19	A	Fuel Element Outlet Gas Temperature Indicator
19	B	Fuel Element and Outlet Gas Thermocouple Patch Panel
C-1	----	Intercom and Emergency Signals
C-2	----	Shutdown and Emergency Cooling Controls
C-3	----	Interlock Indicators
C-4	Upper	Nuclear Instrument Indicators
C-4	Lower	Safety and Shutdown Rod Controls
C-5	----	Main Console Power Switch and Control Circuit Indicators
C-6	Upper	Nuclear Instrument Indicators
C-6	Lower	Shim, Setback and Fine Rod Controls
C-7	----	Log Count Rate Detector Chamber Shield Operating Controls
C-8	----	Main Coolant Loop Parameter Controls and Instruments
C-9	----	Temperature Display Oscillograph

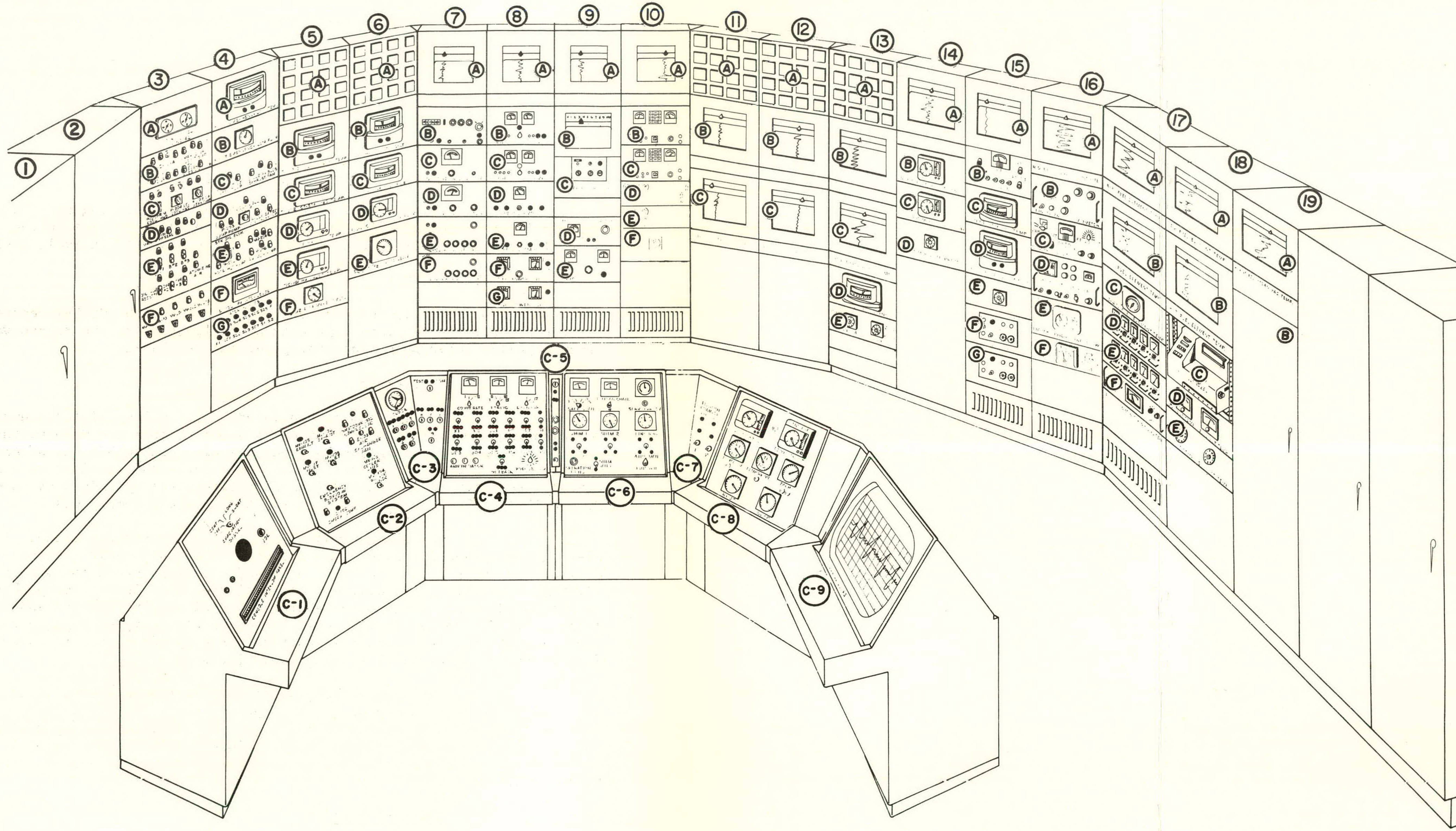

FIGURE IV-9. GCRE CONTROL ROOM LAYOUT

See legend on page IV-16

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5. Emergency Controls

Control of all equipment used to avert or correct an emergency situation was possible from the control room. Typical of such functions were actuation of the emergency reactor cooling (blowdown) system, actuation of the emergency water system pump for fire fighting, actuation of evacuation and air raid alarms, actuation of water valves to flood the mechanical equipment pit, etc. A multichannel radiation monitoring system provided information concerning radiation levels at selected locations throughout the facility.

6. Interlock, Setback and Scram Circuits

This control circuitry was provided to augment the administrative controls contained in the operating procedures to assure the safe operation of the reactor. These circuits are summarized in Table IV-3.

E. FUEL ELEMENTS

Two fuel element concepts were tested in the GCRE-I reactor. The first was a dispersion plate-type element designated IZ; the second was a ceramic pellet loaded pin-type concept designated IB-2L.

1. IZ Element

The IZ element consisted of 315 $\pm$ 5 gm of fully enriched UO_2 dispersed in AISI Type 316 stainless steel to form a 0.033-in. thick "meat" section of 30 wt% UO_2. The meat was clad with 0.006 in. of AISI Type 318 stainless steel. The plates were formed into 120^o sections of a right circular cylinder and assembled into four concentric cylinders as shown in Figure IV-10. The active length of the fuel plate assembly was 28 in.

The element was assembled about a central spine. The fuel plate assembly support spider was pinned to this spine near the top of the element and the plates were free to expand downward into grooves in the bottom fitting. The fuel plate assembly was enclosed in an insulation package which consisted of an inner and outer stainless steel sheath with the annular space filled with Thermoflex insulation. Appropriately designed top (to support the fuel element from the calandria upper tube sheet) and bottom (to provide for a flow distributing orifice) fittings were included.

The element hung vertically in the calandria pressure tube. In operation, gas flowed downward through the spaces on either side of the four fuel cylinders. With 800^oF inlet gas, the maximum (hot spot) fuel plate surface temperature was calculated to be 1650^oF. The nominal average cladding temperature was 1350^oF. The design data for the element is summarized in Table IV-4.

2. IB-2L Element

The IB-2L fuel element consisted of an array of metal clad fuel pellets in an insulation package similar to that used in the IZ design. The fuel was fully enriched UO_2. This material was cold-pressed and sintered to about 96% of theoretical density for loading in six of the fuel pins.

TABLE IV-3

GCRE INTERLOCKS, SETBACKS AND SCRAMS

FUNCTIONS	ACTIVATION CONDITION	BY-PASSABLE (1)
Interlocks		
Permissive for initiation of rod withdrawal	All rods full in*	No
" " "	Detectable source count rate	No
" " "	Reactor inlet coolant temperature at specified value	Yes
" " "	Reactor pool full	Yes
" " "	Demineralized cooling water system in operation	Yes
" " "	Cooling tower circulating pump in operation	Yes
" " "	Coolant system pressure at specified value	Yes
Disconnect high voltage from Channels I & II Detectors	Channels III & IV reading >5% power	No
Setbacks (2)		
Reduce reactor power	Coolant flow <90% of specified value	No
" " "	Reactor power >110% of specified value	No
" " "	Period (Channels III & IV) <10 sec	No
" " "	Reactor outlet coolant temperature more than 40°F above specified value	No
" " "	Manual	No
Scrams (3)		
Shutdown reactor	"Run-safe" switch (any one of five) in "Safe" position	No
" "	Coolant flow <85% of specified value	No
" "	Log-N >115% of specified value	Yes
" "	Period <7 sec	Yes
" "	Earthquake (seismic) switch open	No
" "	Safety rod scram pressure less than specified value	No
" "	Reactor outlet coolant temperature more than 60°F above specified value	No
" "	Failure of normal electrical power	No
" "	Reactor power >115% of specified value (Channel V, VI, VII or VIII)	No
" "	Manual	No

* Withdrawal sequence interlocks were also provided; these required that all safety and shutdown rods be fully withdrawn before any other rod could be withdrawn, that the setback rod be withdrawn before the fine or shim rods could be withdrawn and that the shim rods be moved off the "in" limit before the fine rod could be withdrawn.

(1) Items indicated "Yes" could be deactivated with key-locked switch; items marked "No" could be deactivated only by altering circuitry.

(2) Automatic insertion of setback rod reducing reactor power 30% in 3 seconds and 70% in 90 seconds.

(3) Automatic insertion of safety, shutdown, shim, setback and fine rods.

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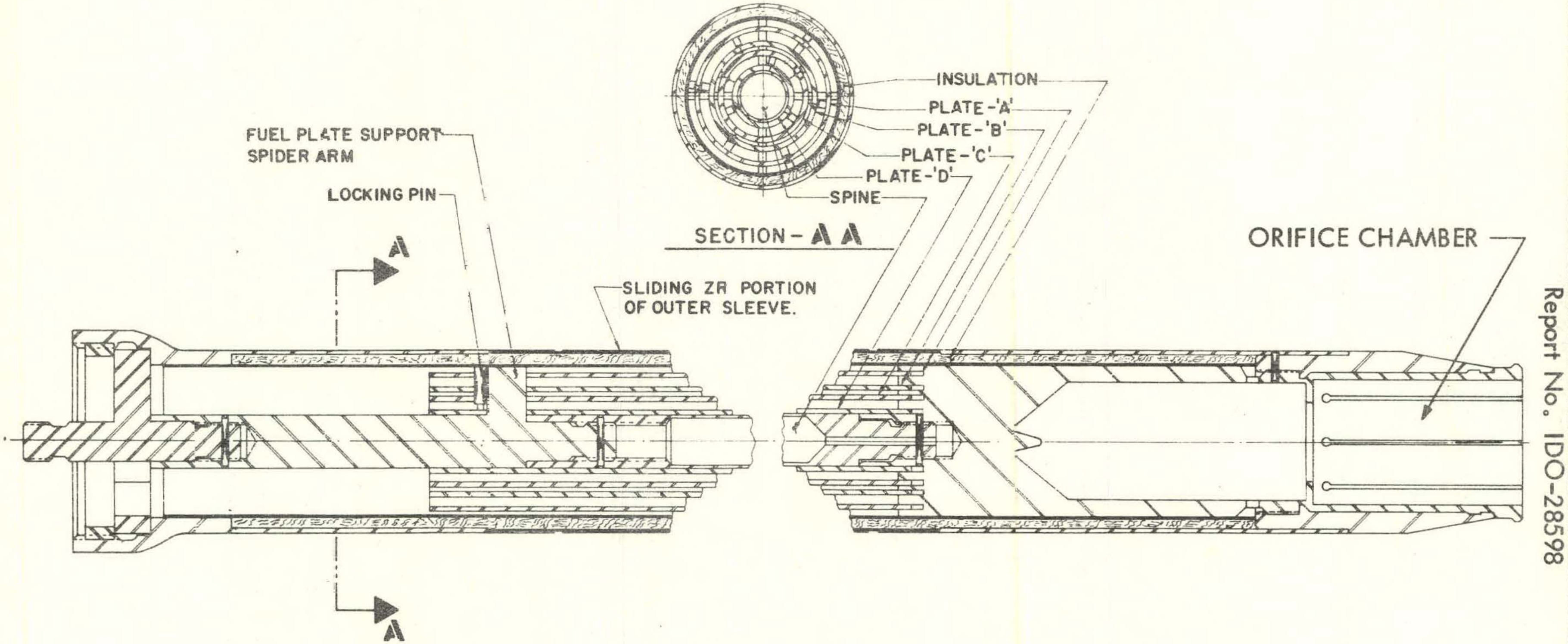

FIGURE IV-10. GCRE-I PLATE-TYPE FUEL ELEMENT

TABLE IV-4

IZ FUEL ELEMENT DESIGN DATA

Total length, in.	44
Active length, in.	28
Outside diameter, in.	1.72
Fuel material	Fully enriched UO_2
Fueled section	UO_2 dispersion in AISI Type 316 stainless steel
Fuel loading, wt%	30
Fuel loading, gm (nominal)	315.0
Cladding material	AISI Type 318 stainless steel
Cladding thickness, in.	0.006
Central spine	
Material	AISI Type 316 Stainless Steel
OD, in.	0.499
Wall thickness, in.	0.032
Inner liner	
Material	AISI Type 316 stainless steel
OD, in.	1.446
Wall thickness, in.	0.010
Insulation	
Material	Thermoflex
Thickness, in.	0.215
Outer liner	
Material	Reactor grade zirconium
OD, in.	1.720
Wall thickness, in.	0.030
Burnable poison	
Location	Inside of outer liner, center of active length
Material	Samarium
Quantity, gm	0.163
Configuration	0.005-in. thick foil, 6 in. long

For the remaining 12 fuel pins (the central pin was unfueled), the fuel pellets were fabricated from a mixture of 70 wt% fully enriched UO_2 and 30 wt% BeO. The fuel pellets were loaded into Hastelloy X tubing which had been fitted with top and bottom closure plugs. Eighteen loaded fuel pins were arranged in two concentric rings about the unloaded central pin to form the fuel pin array. This assembly was supported from the top by a spider and positioned by a similar device at the bottom which provided for thermal growth of the pins in the downward direction. The insulation package consisted of AISI Type 316 stainless steel inner and outer liners; the void between the liners was filled with Thermoflex insulation. A burnable poison foil of cadmium alloy was attached to the inner surface of the outer liner. A bell housing at the top of the element supported the fuel pin spider, provided a seating surface for the fuel element in the calandria upper tube sheet and provided a point of attachment for the insulation package. A suitably designed bottom fitting incorporating an orifice was provided. Spacer wires were wrapped around the fuel pins to maintain a gas flow passage.

A drawing of the IB-2L element is shown in Figure IV-11 and a photograph of the top portion of the fuel pin bundle is presented in Figure IV-12. The design data for the IB-2L element is tabulated in Table IV-5.

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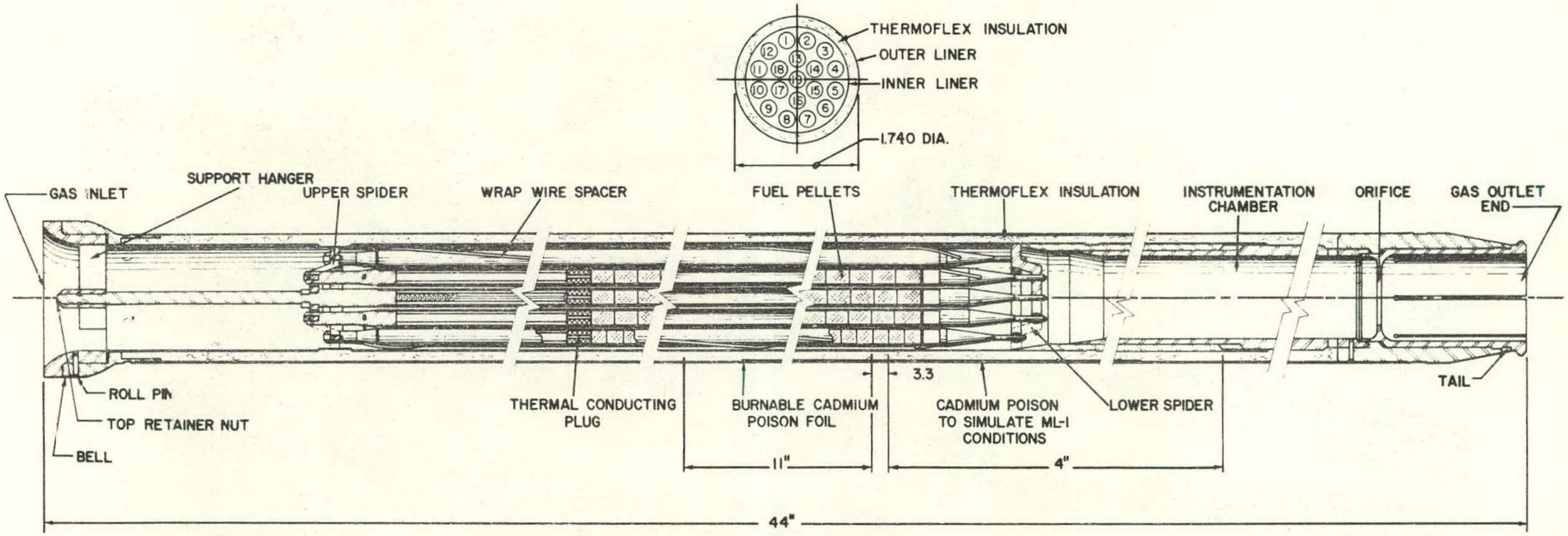

FIGURE IV-11. GCRE-I PIN-TYPE FUEL ELEMENT

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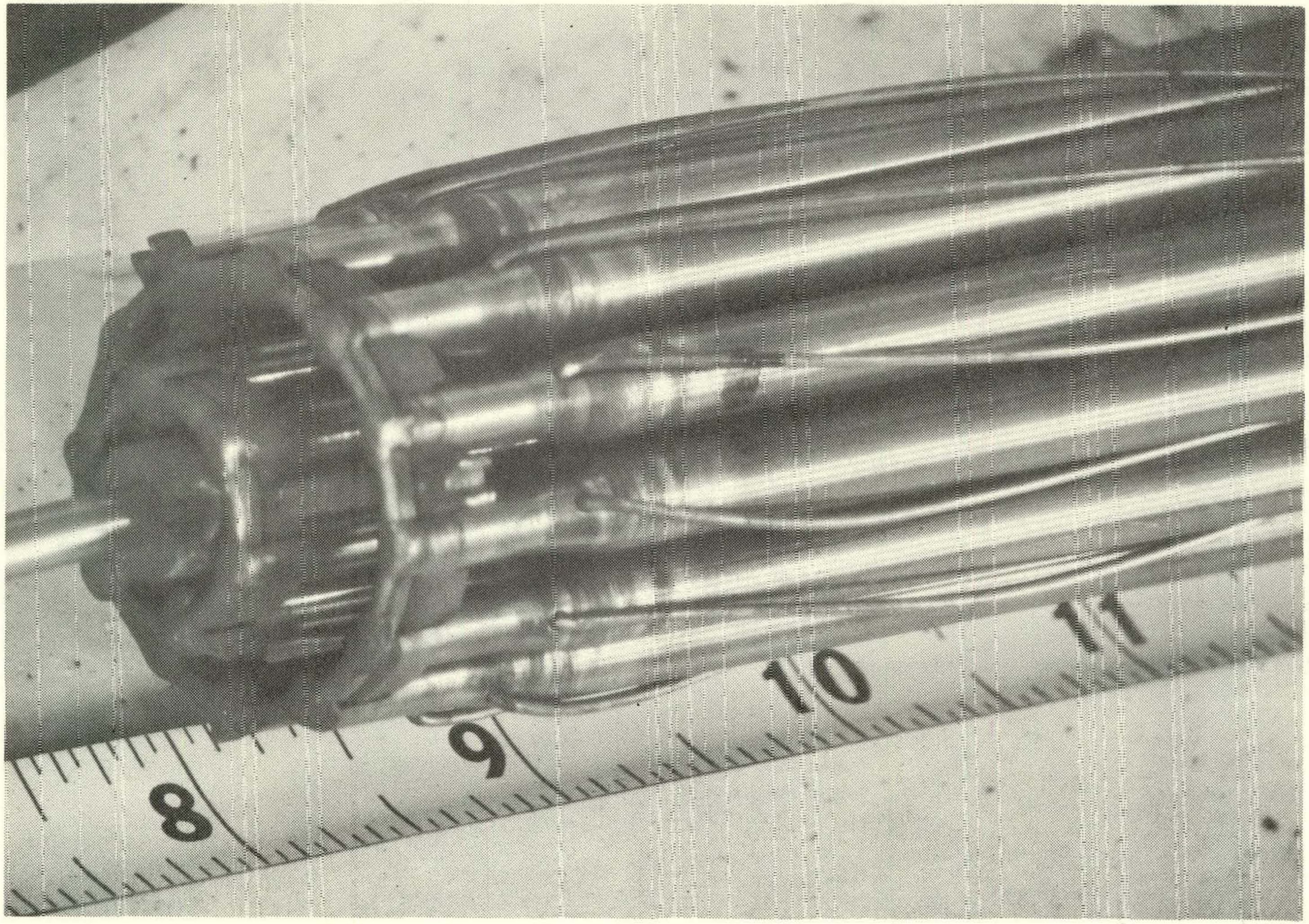

FIGURE IV-12. PIN-TYPE FUEL ELEMENT SHOWING SPIDER AND PIN ASSEMBLY

TABLE IV-5

IB-2L FUEL ELEMENT DESIGN DATA

Total length, in.	44
Active length, in.	22
Outside diameter, in.	1.740
Fuel composition, inner pins	Fully enriched UO_2
outer pins	60 wt% fully enriched UO_2; 40 wt% BeO
Fuel loading, gm	800
Fuel configuration	Cold pressed, sintered pellets
Pellet diameter, in.	0.175
Length, in.	
Cladding	
Material	Hastelloy X ($\sim$ 1% Co)
OD, in.	0.241
Wall thickness, in.	0.030
Inner liner	
Material	AISI Type 316 stainless steel
OD, in.	1.446
Wall thickness, in.	0.010
Insulation	
Material	Thermoflex
Thickness, in.	0.28
Outer liner	
Material	AISI Type 316 stainless steel
OD, in.	1.740
Wall thickness, in.	0.010
Burnable poison	
Material	Cadmium
Configuration	Foil; 11 in. long on inside of outer liner; top edge 3.85 in. above center of fuel
Spacer wire	
Material	Hastelloy X
Diameter, in.	0.040

V. OPERATIONAL HISTORY

This portion of the report discusses the significant occurrences during the startup and test operation of the GCRE facility and the GCRE-I test reactor. The subjects are presented by equipment categories rather than as a chronology (although major items are placed in time) to facilitate understanding of the major operational problems and accomplishments.

A. FACILITIES AND PROCESS EQUIPMENT

The construction contractor completed work and turned the GCRE facility over to Aerojet in early October 1959. In the period which followed, AGN technicians checked out and test-operated all facility equipment concurrent with the completion of installation of the reactor and instrumentation and controls systems. The significant occurrences associated with this work are summarized below.

1. Main Compressor

As anticipated, some leakage through the water-buffered seal was observed during the initial operation of the compressor but this leakage soon disappeared as the seal was worn in. The compressor operated without incident for the balance of the GCRE-I operating program. Minor modifications were performed to improve the operating characteristics of the lubrication and seal water system.

After initial checkout and alignment, the eddy current speed-controller operated without significant problems throughout the GCRE test period. The cooling water supply was modified early in the program to supply utility water rather than water from the regular equipment-cooling system because of the undesirable effects of the water-conditioning additives in the latter system. In addition, the water return system was modified to provide a drain tank with an automatically controlled pump to eliminate the possibility of pressurization of the cooling circuit in the speed controller.

2. Oil-Fired Heater

Some difficulty was experienced with this unit. The major problem was associated with excessive temperatures in the heater pit which resulted in failure of electrical insulation on the control wiring. Over a period of several months, the control wiring was replaced with material of higher temperature resistance, the solenoid valves which controlled oil flow to the burners were replaced with more rugged and dependable units, and louvers were installed in the building wall to improve the ventilation in the heater pit. Although these modifications resulted in temperatures which were satisfactory for heater operation, it was necessary to provide an insulated and ventilated enclosure for the flow metering instrumentation located in the heater room to minimize the effects of the high ambient temperature on that system.

After several hundred hours of operation, a routine inspection revealed that the upper support for several of the gas tubing passes in the heater had failed. After consultation with the vendor, the tubes were jacked back into position and anchored with a support of improved design. Complete inspections, including radiographic examination of the support and the highly stressed areas of the tubes, indicated that no damage had occurred. There was no evidence of any further problem during the GCRE test program.

3. Shutdown Cooling Loop

A minor modification was made to the gas relief piping system to prevent the failure of the burst diaphragm (which protected the shutdown compressor from over-pressurization) during normal pressure surges associated with the startup and shutdown of the main coolant loop.

4. Gas-to-Water Heat Exchanger

This unit operated without incident throughout the test program. Prior to reactor startup, a test was performed to determine the capability of the exchanger to operate under non-standard conditions. During this test, the external loop was operated in a fashion which resulted in a heat transfer rate of 1×10^6 Btu/hr at an inlet temperature to the exchanger of 690^{o}F. Under these conditions circulation of demineralized water was discontinued and it was observed that the gas temperature at the outlet of the heat exchanger stabilized at 170^{o}F which is about 30^{o}F higher than normal. With the demineralizer loop circulating, the flow in the cooling water system was discontinued. Under these conditions, the gas temperature at the outlet of the gas-to-water heat exchanger rose from 110^{o}F to 182^{o}F in about 20 minutes. These tests demonstrated the adequacy of the heat dump system to prevent serious over-temperatures in the event of a power failure which resulted in loss of demineralized or cooling water flow.

5. Gas Filtering Equipment

The method of attachment of the covers on the main gas system filters proved unsatisfactory during the initial checkout. A field modification was made which provided a gas-tight closure and, at the same time, made possible remote removal of the covers.

The filter elements were not changed during the GCRE test program. Following the final power operation, the elements were removed and it was observed that the activities were quite low. Spectrographic analysis of the samples of the material identified small amounts of Co-58 and Cr-51.

6. Oxygen System

The original design of the gas make-up system provided for the addition of oxygen in the required amounts from a supply of pure oxygen. Repeated efforts to place this system in satisfactory operation were unsuccessful and, prior to initial startup of the reactor, the concept of direct addition of oxygen to the main coolant loop was abandoned in favor of a procedure which pre-mixed oxygen and nitrogen in the bulk nitrogen storage tanks.

7. Demineralized Water System

This system operated essentially without incident throughout the test program. A piping revision was made which permitted water circulation through the 50-gpm demineralizer for cleanup prior to the introduction of demineralized water into the pool or other components. In addition, a commercial water softener was installed up-stream of the demineralizers to reduce the load and increase the cycle lifetime of these units.

8. Reactor Pool

During early phases of the operation, it was observed that atmospheric dust collected on the surface of the pool to an extent which was considered undesirable for operation. A skimmer system was installed which collected surface water for recirculation through a filter to the demineralizer. This system was effective during normal operation; during periods when the operating program would permit, a plastic cover was installed on the pool to further control the contamination of the water.

9. Cooling Tower

A significant modification was made to the circulating piping at the cooling tower to prevent ice formation during freezing weather when no heat load was applied to the system. This modification involved a valved bypass line which permitted bypass of the cooling tower and recirculation of water directly to the tower basin. This arrangement functioned satisfactorily and eliminated all problems of ice buildup in the cooling tower.

10. Make-up Nitrogen Storage

The original design for this system made provision for the connection by manifolding of more than 100 standard bottles of nitrogen for system make-up and emergency blowdown. It was apparent early in the program that the work load and safety hazards associated with the handling of so many bottles were unacceptable and, in cooperation with the Linde Company, a bulk storage facility was provided. Under the new arrangement, high pressure bulk storage cylinders provided make-up gas and a source for charging a relatively small number of permanently installed standard bottles which were retained for the emergency blowdown system. This modification proved completely acceptable and the system was simple and safe to operate.

11. Pressure Control Valves

The pressure control valves provided in the system were of a new design which had been subjected to very limited field service. Considerable difficulty was experienced with these valves during the pre-startup checkouts and it became necessary to remove the valves and to install units which had been proven reliable in similar applications.

12. Utility Water Well

Shortly after activation of the well which provided all the water for the GCRE facility, a routine laboratory test of the water revealed the presence of an unacceptably high concentration of organic material. Special provisions were made for purification of the drinking water supply and for isolation of this water system from all others in the building. Subsequent checks of the organic concentration in the water indicated a gradual decline in the contamination level; after a period of approximately one year, the water was considered acceptable for all uses without special treatment.

13. Liquid Waste Handling System

It became apparent early in the checkout program that large quantities of potentially radioactive liquid waste, containing very low levels of contamination, would be generated in the normal operation of the facility. In view of the limited capacity of the low level liquid waste storage tank, provision was made for gravity transfer of material from this tank (after determination of the radioactive content) to a leaching bed west of the GCRE facility. This modification involved the installation of several hundred feet of 6-in. concrete pipe along the north side of the test building. Operating procedures were established which specified that the liquid waste in the storage tank be monitored prior to release to the bed and that the bed be monitored on a periodic basis to prevent buildup of dangerous levels of contamination. No problems were experienced with this system throughout the program.

14. Emergency Diesel Generator

The only significant problem encountered with this unit resulted from improper insulation of the exhaust stack at the point of penetration of the building roof. After several hours of operation, the temperature at this point reached the level which ignited the bituminous material on the roof, resulting in a small fire. An insulating sleeve was installed around the exhaust duct and no further problems were encountered.

In summary, the checkout of the facility for integrated operation with the reactor proceeded expeditiously and smoothly. Although several significant modifications (as described above) were required, problems with the facility and process equipment did not contribute significantly to the delays experienced in the test program; these delays were associated almost exclusively with the control rod system and the reactor calandria as described below.

B. REACTOR AND CONTROL SYSTEM

The significant occurrences associated with the startup and operation of the reactor and control system are summarized below.

1. Calandria

As discussed earlier, it was necessary to replace the aluminum calandria originally provided for the GCRE reactor prior to initiation of power operation. As indicated in the description of the stainless steel calandria, internal cooling passages were provided in the tube sheets to reduce the temperature and associated thermal stresses in these thick sections.

In April 1961, after the reactor had operated satisfactorily for more than 2000 Mw-hr, gas leakagé was observed in the area of the upper tube sheet coolant inlet manifold and discharge ports. The reactor was immediately shut down and a check of support system conditions revealed that the tube sheet coolant circulating pump was not operating because of an open circuit breaker in the electrical supply line. The pump was started and it was observed that the gas leakage was confined to a single discharge port (see Figure V-1). Subsequent borescopic inspection of the tubes in the calandria revealed a horizontal crack at the lower edge of the coolant channel area in a single tube. This crack extended approximately 120° around the inner circumference of the tube as shown in Figure V-2.

While arrangements were being made to remove the section of the failed tube for metallographic evaluation and to plug the tube to permit resumption of reactor operation, an extensive program of inspection of the calandria was undertaken. A detailed borescopic examination of all pressure tubes in the calandria indicated that no additional cracks existed. Several suspicious appearing areas were observed but, in each case, the appearance of the tube was ultimately related to the as-received condition of the tubing or the results of fabrication techniques used in the assembly of the calandria.

A section of the failed tube was removed and a plug installed in both ends of the sectioned tube. The metallurgical analysis of the removed section indicated that the tube had failed as a result of corrosion fatigue initiated from the outside (coolant channel) surfaces of the tube. Calcium deposits on the tube indicated that boiling had occurred in the cooling passages as a result of the lack of cooling water circulation (Figure V-3).

At the conclusion of the sectioning and tube plugging work, a non-nuclear thermal test of the reactor was conducted. In this test the reactor was subjected to several heating and cooling cycles by varying the temperature of the gas. During the test, numerous leaks were observed in the area of the upper tube sheet and it was concluded that the calandria had deteriorated to the point where continued operation or repair was impossible. The subsequent disassembly and destructive examination of the calandria supported the earlier conclusion that the failures were the result of boiling in the upper tube sheet coolant passages which accelerated corrosive processes and led to the ultimate fatigue failure of the tubing. This subject is discussed fully in the final published report of the incident (Ref. 11).

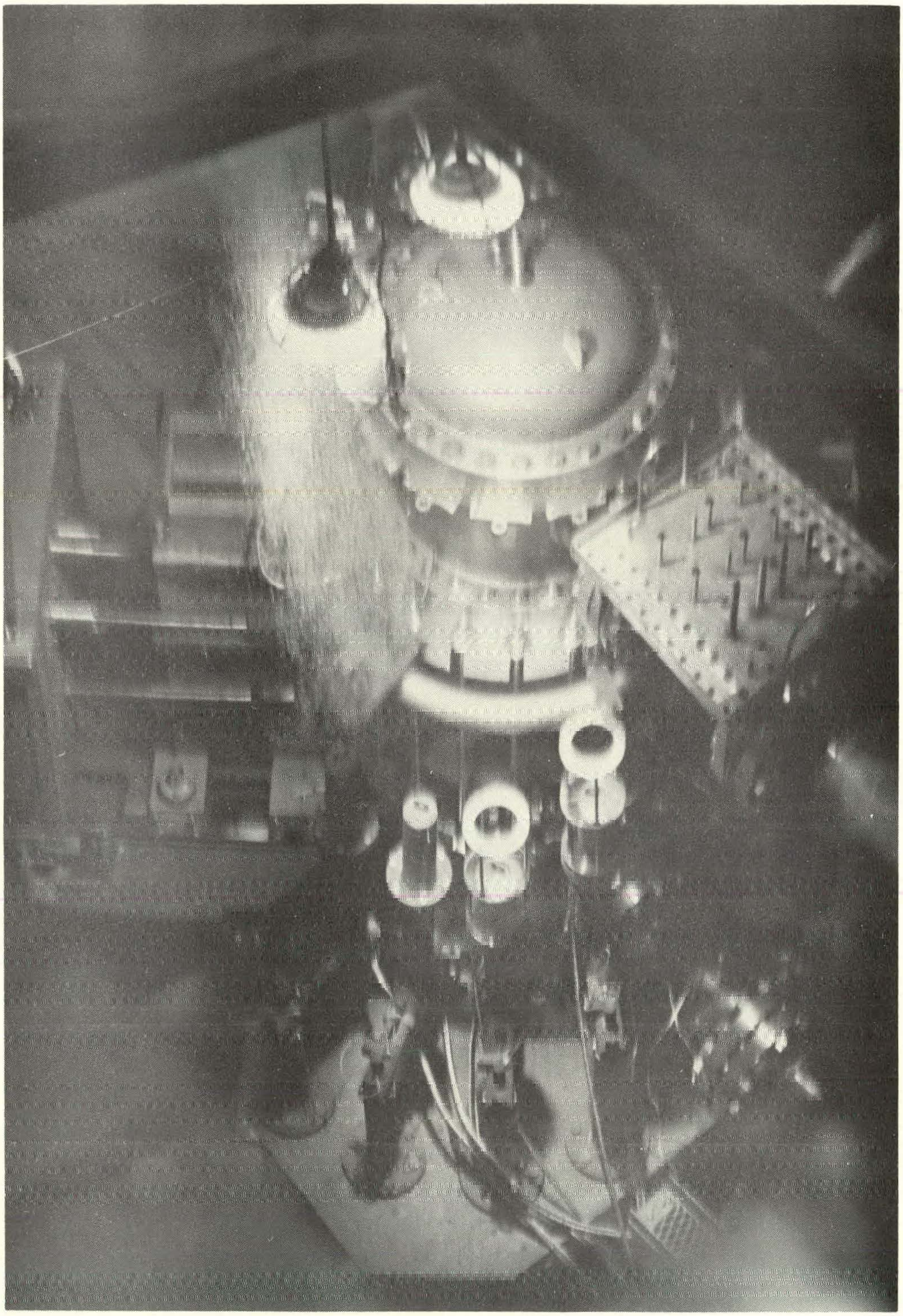

FIGURE V-1. GAS LEAK IN GCRE-I STAINLESS STEEL CALANDRIA

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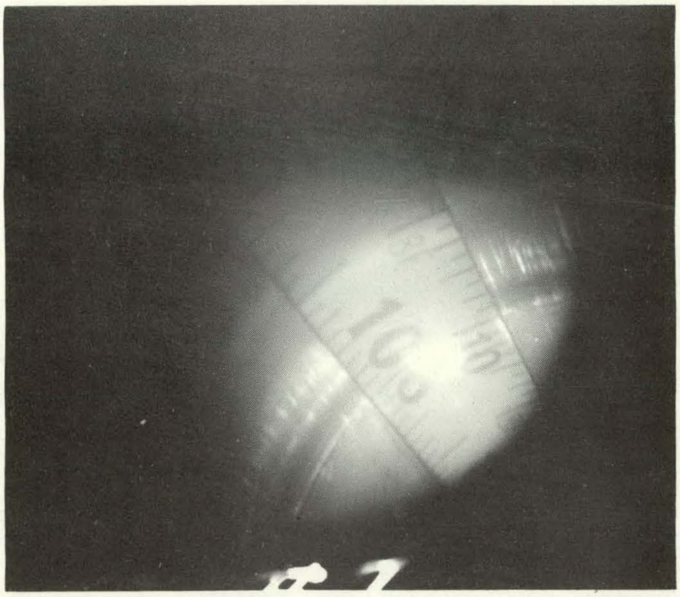

FIGURE V-2. CRACK IN PRESSURE TUBE, GCRE-I STAINLESS STEEL CALANDRIA

FIGURE V-3. EXTERIOR SURFACE OF PRESSURE TUBE SECTION

(Note the white calcium deposit on the upper portion of the specimen. Deposit is at top of water coolant passage)

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2. Plenum Flange Leakage

During the initial checkout of the reactor, leakage of coolant gas was detected at the three main flanges in the reactor. Investigation determined that the flange closures were improperly designed. The reactor was modified by providing serrations in the lands and grooves of the mating flange parts and installing a soft aluminum gasket. This design performed satisfactorily during the initial phases of testing.

Following a significant period of full power operation which included several temperature cycles, leaks were again detected in the lower main flanges of the pressure vessel. It was shown that the leak rate was sensitive to temperature; maximum leakage occurred at 600°F and the leak rate was reduced significantly at the normal (1200°F) operating temperature.

The reactor had not been designed for remote maintenance of the type required to correct the leakage and the radioactivity of the reactor and immediately associated components prohibited contact maintenance. An evaluation of methods of correction indicated that the development of remote operating equipment to tighten the bolts, while feasible, would be extremely expensive and time-consuming. As an alternative, the decision was made to perform the work with divers working approximately 5 ft from the reactor. Tooling and equipment were developed and tested with an underwater mock-up of the reactor components involved. The work was then undertaken on the reactor and, after installation of nut retainers, the bolts in both the lower and upper main flanges were torqued to 2400 ft-lb. Figure V-4 shows a diver engaged in a typical bolt-tightening operation. In the course of this work, the two divers involved received total body exposures of 600 mr and 450 mr respectively. Thermal cycling tests performed at the conclusion of the bolt tightening revealed that the work had been effective.

Near the end of the GCRE-I operating period, minor leakage was again observed at the flanges. In this instance, it was possible, as the result of modifications made during the underwater tightening operation, to re-torque the flange bolts with remote operating tooling from the work bridge over the reactor pool.

3. Nuclear Instrumentation

During the pre-startup preparations, the GCRE nuclear instrumentation did not perform satisfactorily. The primary problem was associated with noise in the Log-N system which resulted in spurious actuation of the period scram circuits. A comprehensive program for correction of this condition was undertaken. Redesigned ion chamber power supplies were installed which regulated the voltage to the ion chamber and eliminated this source of noise in the system. The signal leads from the chambers to the detection chassis were rerouted and shielded from electromagnetic pickup. The instrumentation grounding system was revised to provide an isolated common ground for the nuclear instrumentation system. As a result of this activity, the performance of the nuclear instruments during reactor operation was satisfactory and the number of spurious scrams experienced was comparable to that considered normal for experimental reactor installations.

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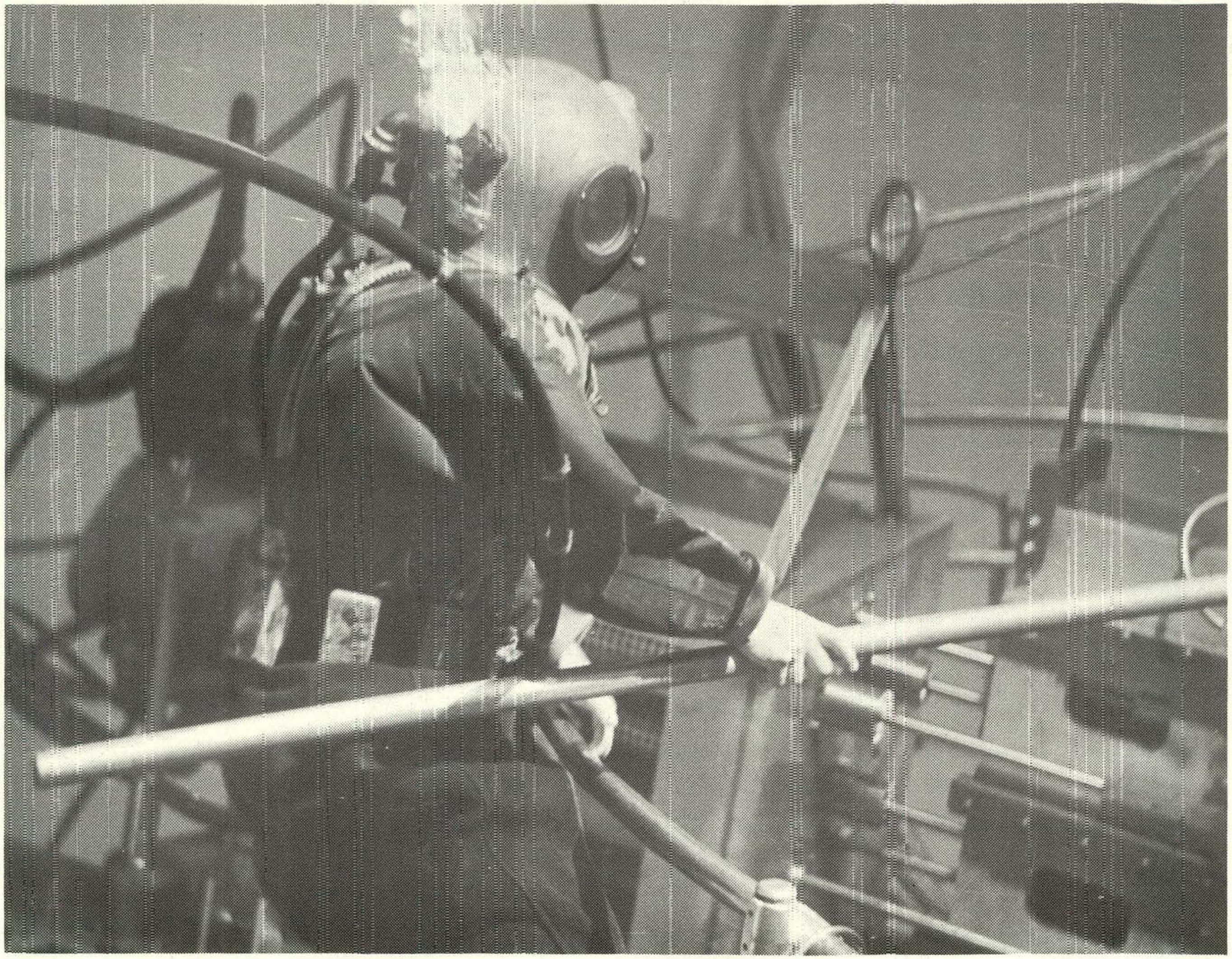

FIGURE V-4. GCRE-I FLANGE BOLT TIGHTENING OPERATION

Showing the diver engaged in a typical bolt tightening operation. Note the temporary restraining handrail to control approach distance (and, thus, exposure) to the reactor.

4. Control Rods

Despite the extensive development program conducted prior to the installation of the control rods at the GCRE facility, the pre-start-up checks revealed that the fast-acting rods did not perform satisfactorily. The major problems observed were intolerably long scram times resulting from the drag forces imposed on the pistons by the seals and the failure of the electromagnetic latches to consistently hold the fast-acting portion of the rod assembly. The decision was made to perform expedient modifications to the actuators and to proceed with low power testing with the aluminum tube bundle. Concurrently, the basic developmental program was continued to produce a satisfactory design and it was contemplated that the necessary modifications would be made at the time of installation of the stainless steel tube bundle. The expedient modifications permitted acceptable operation of the reactor during the low-power testing in February and March of 1960. Following this testing, the reactor was shut down for replacement of the tube bundle and the control rod actuators were modified by the substitution of a graphite-impregnated rubber U-cup seal for the chevron-type seals installed originally and by the replacement of the magnets in the latches with new units with a significantly greater holding force. This modification program as well as the several minor changes made to the actuators during the balance of the GCRE operating period are fully described in the final report on this subject (Ref. 12).

5. Fuel Elements

In early 1961,during a routine inspection of the pin-type fuel elements,(while the reactor was shut down for other maintenance work) it was revealed that significant damage had occurred in the upper areas of the elements. In several cases, the retaining nut on the central support shaft had been sheared off at the point of contact with the upper support spider (see Figure V-5 and V-6) and in almost every case the pins which joined the pin bundle assembly to the insulation liner assembly had been sheared. These failures had permitted the hanger assembly to become separated from the upper bell housing (see Figure V-7) which raised the fuel pin bundle sufficiently so that, in some elements, the fuel pins had withdrawn from the lower spider.

Laboratory tests were conducted to determine the mechanism of failure and to define the corrective modification. These tests revealed that the shearing of the pins was the result of high frictional forces in the expansion joint area caused by excessive insulation in that area. The failure of the expansion joint to operate properly had resulted in thermal ratcheting which developed forces sufficiently great to shear the pins. The failure of the central support rod was attributed to impact loading of the element during normal fuel element handling.

A cylindrical support was designed to replace the support rod between the hanger and the upper spider. This unit was fabricated from Hastelloy X and installed remotely by a simple insertion and rotation action. The new support eliminated the requirements for the nut on the central support rod and the nut was subsequently removed from all the IB-2L fuel elements. In addition, the support was designed to lower the pin bundle about 1/8 in. and assure that further thermal expansion did not result in the pin tips being

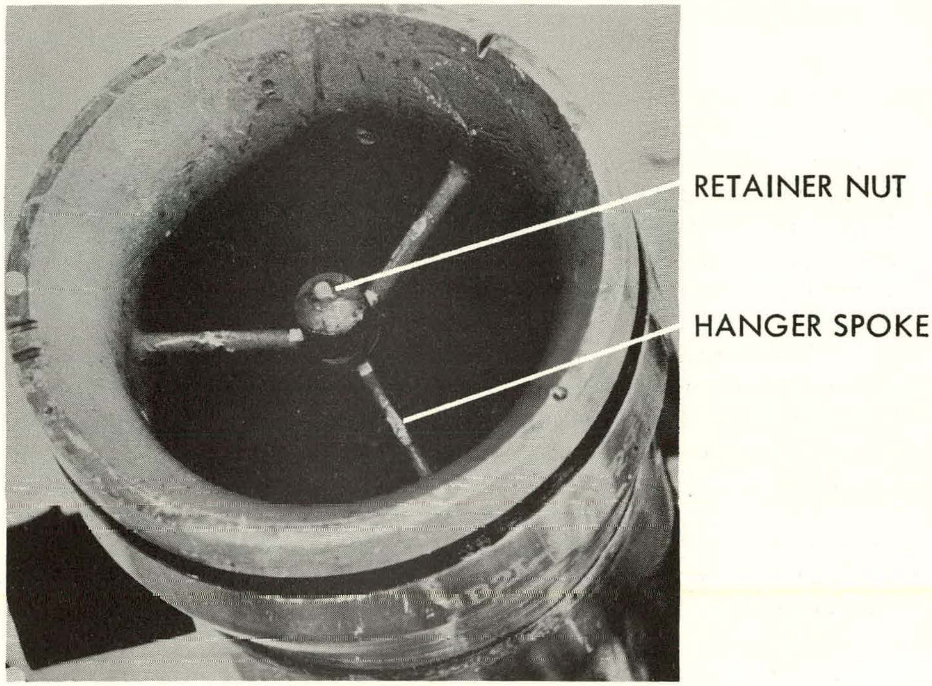

FIGURE V-5. IB-2L FUEL ELEMENT MALFUNCTION

Note central support shaft where retaining nut has been sheared off.

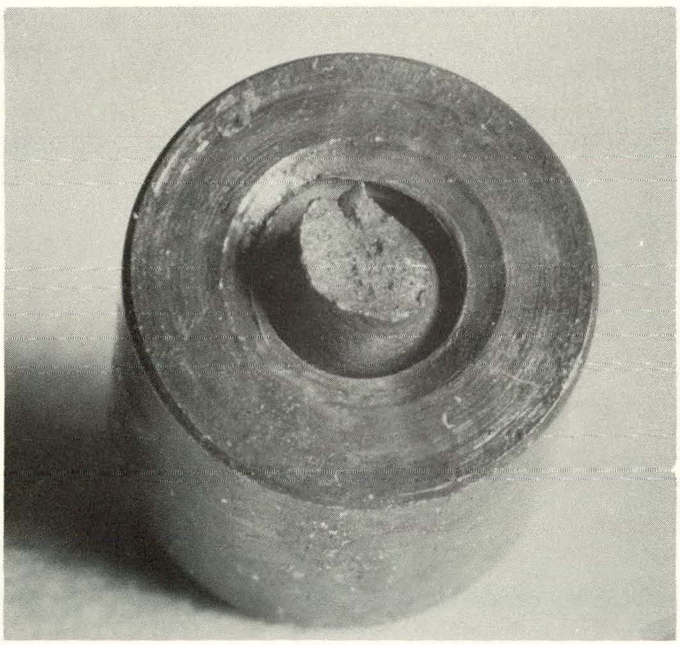

FIGURE V-6. IB-2L RETAINING NUT

This nut was recovered from the GCRE-I reactor after shearing off the central support shaft.

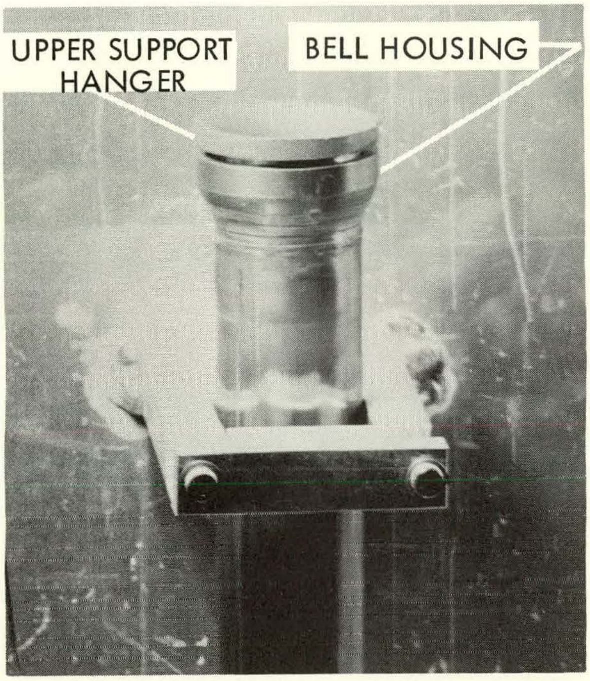

FIGURE V-7. IB-2L FUEL ELEMENT SEPARATION

Showing the separation of the upper support hanger from the bell housing.

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pulled out of the lower spider. Those connecting pins between the hanger and the bell support which had not sheared as a result of reactor operation were deliberately sheared to prevent further malfunction of the expansion joint. The fuel elements performed satisfactorily and no further evidence of damage was observed either during operation or at the time of removal at the conclusion of operation.

VI. EXPERIMENTAL PROGRAM

The experimental program conducted with the GCRE-I had as the principal goal the generation of neutronic and engineering data in support of the design and development of the ML-1 reactor.* In addition, the program included an initial phase consisting of a series of experiments to determine the neutronic characteristics of the GCRE-I; this information was required for the safe operation of the reactor during the conduct of subsequent tests. Each experiment was conducted in accordance with an approved detailed procedure; a typical experimental procedure is shown in Appendix E.

In this section, the various experiments are discussed in chronological order. In each discussion, the purpose of the experiment, the method of conduct and the significant experimental results are presented; where appropriate, the significance of the results in terms of GCRE-I evaluation or ML-1 support is discussed.

A. EXPERIMENTS WITH PLATE-TYPE (IZ) FUEL ELEMENTS IN THE ALUMINUM CALANDRIA

As indicated in Section IV-A, irreparable leakage in the aluminum calandria resulted in the decision to fabricate and install a stainless steel assembly in the reactor. During the period when the design and fabrication of the replacement calandria was in progress, several low-power experiments were conducted with the plate-type fuel elements in the aluminum unit. The purpose of these tests was to develop preliminary estimates of the neutronic characteristics of the reactor, to evaluate procedures for subsequent use and to train the operating crew and accumulate operating experience. Although the former goal would not contribute directly to the ML-1 effort (since the fuel elements and calandria were both "non-prototype" for ML-1), it was felt that the data obtained could reveal significant problems, if such existed, in the concept of a closed-cycle, gas-cooled, nuclear energy source. In addition, the completion of the latter goals would certainly improve the performance of the crew during the later, more meaningful testing.

The experiments undertaken are discussed below.

*The energy source for the demonstration low-power, gas-cooled nuclear power plant designated ML-1.

1. Initial (Wet) Critical Experiment

The purpose of this experiment was to determine the critical loading of the reactor in the cold, clean condition. The experiment was conducted with the calandria gas passages flooded with water since this represented the most reactive condition of the reactor. The experimental procedure specified the incremental addition of reactivity in the form of fuel elements charged into the core until the loading was less than one fuel element supercritical. A neutron source and sensitive (BF_3) neutron detection instruments were installed in peripheral calandria tubes. Plots of inverse count rate and inverse multiplication factor as a function of the number of fuel elements charged were developed during the experiment. These data were monitored carefully to predict the critical loading and to detect unanticipated conditions.

Criticality was attained with 36 plate-type elements. The estimated excess reactivity was 0.10% $\Delta K/K$ with all control rods full out. This loading corresponded to 11.34 kg of U-235; a critical assembly experiment at Battelle Memorial Institute (BMI) had predicted that 11.318 kg of U-235 would be required. The plots of inverse count rate for the last eight elements charged are shown in Figure VI-1.

2. Full Flooded Core Experiment

The purpose of this experiment was to demonstrate that the operating core loading of 61 elements would be subcritical in the flooded condition with any two control rods withdrawn. Successful completion of this experiment would verify the design criteria that the reactor be safe with any two control rods inoperable.

The core loading was increased from the wet critical loading of 36 elements to the calculated normal operating configuration of 61 elements. Neutron population was monitored by both the special BF_3 instrumentation installed for the wet critical experiment and by the normal startup range nuclear instrumentation (Channels I and II). A plot of inverse multiplication was developed to monitor the progress of the experiment and to detect unanticipated conditions.

When the core consisted of 61 elements loaded in a symmetrical pattern, various control rods, singly and in combination, were withdrawn and the shutdown margin of the core determined from the counting data. By this technique, it was established that the combination of shutdown rods 2 and 5 controlled the greatest amount of reactivity and that, with these two rods withdrawn, the reactor was more than 1.5% $\Delta K/K$ subcritical. Subsequent evaluation of the data indicated that, in addition to shutdown rods 2 and 5, the fine and setback rods could also have been withdrawn without achieving criticality (see Figure VI-2). The experiment demonstrated conclusively that the fully loaded reactor was safe in the most reactive condition (cold, clean and flooded) even though the two strongest control rods were inoperative.

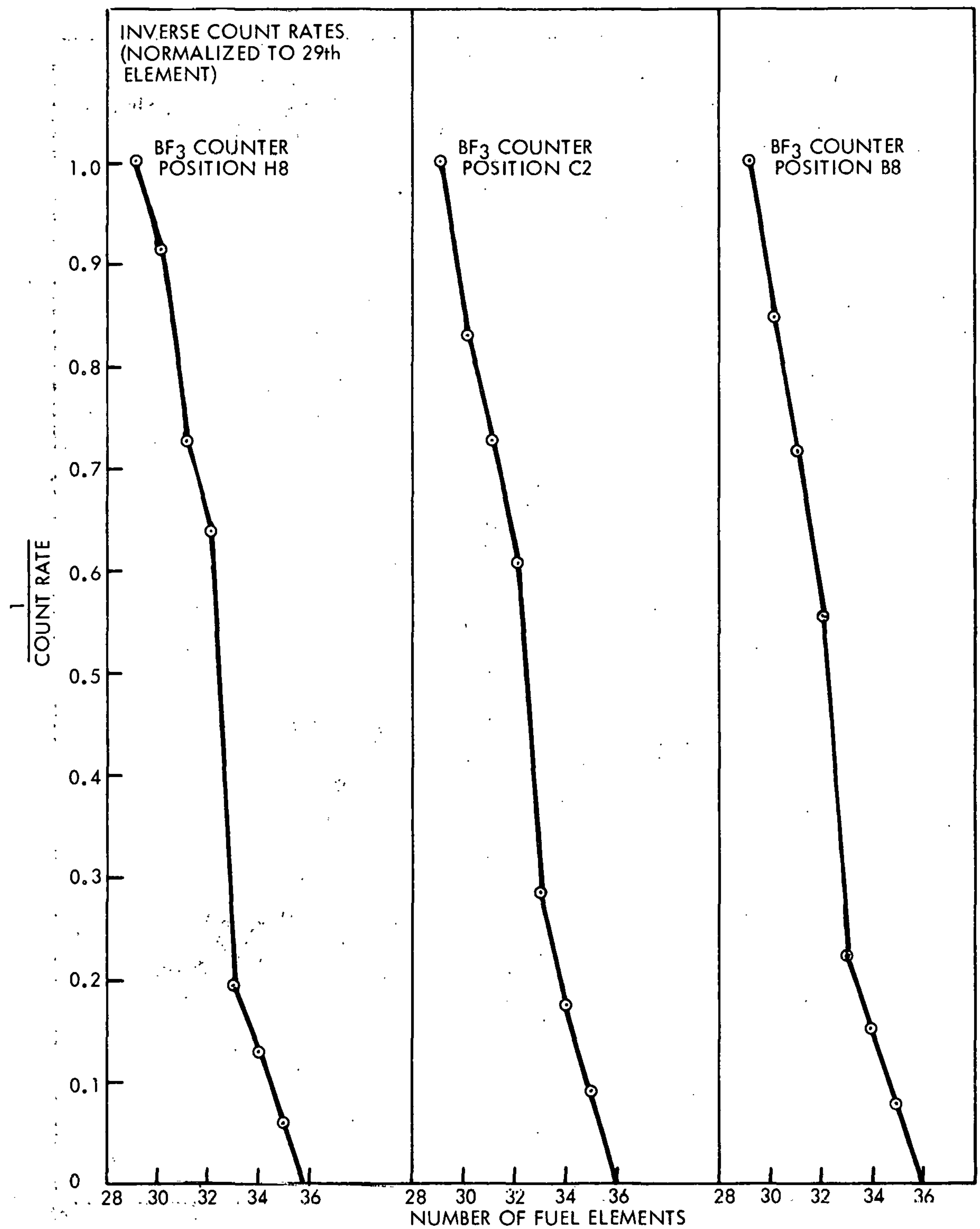

FIGURE VI-1. INVERSE COUNT RATES - GCRE-I INITIAL CRITICAL EXPERIMENT

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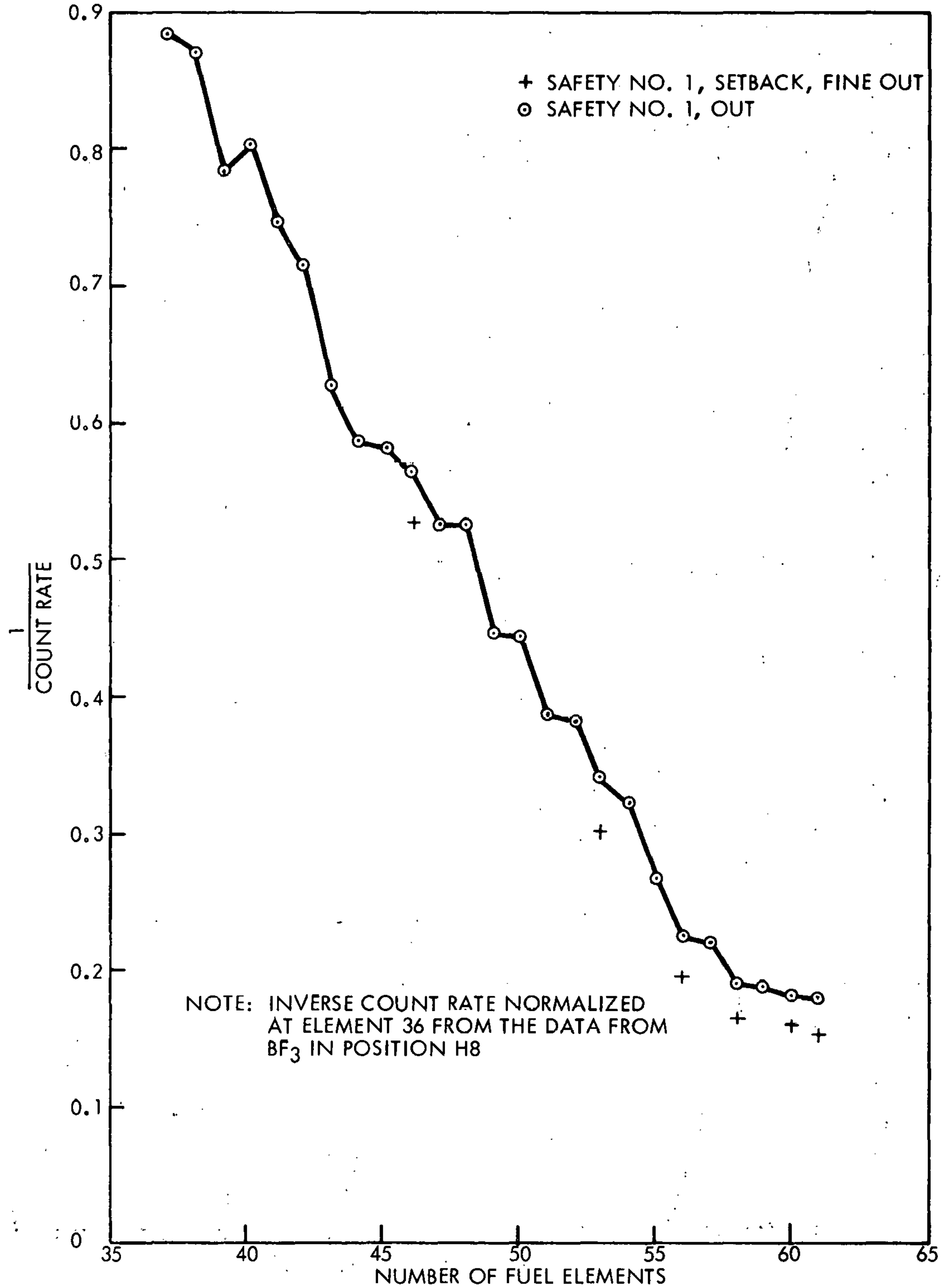

FIGURE VI-2. INVERSE COUNT RATE - GCRE-I FULL FLOODED CORE EXPERIMENT

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3. Reactor Drying Experiment

The purpose of this test was to evaluate the correctness and effectiveness of the reactor drying procedures.

The reactor was drained and gas circulation established with the main compressor. The temperature of the gas was increased from the initial value of 250°F to 600°F by adjusting the operation of the oil-fired heater. During the first 60 hr of operation, moisture condensed in the trap of the gas-to-water heat exchanger and was blown out. The drying was terminated after 84 hr when a dew-point of minus 40°F was indicated in the gas at the reactor outlet. At this point, one fuel element was removed from the reactor and sealed in a closed chamber. A vacuum was drawn on the chamber; no moisture was collected in 24 hr.

On the basis of this experiment, it was concluded that the reactor drying procedures were basically correct and that effective drying was feasible. It was felt that the drying time probably could be reduced in subsequent operation.

4. Dry Critical Experiment

The purpose of this experiment was to determine the loading required to achieve criticality with the reactor in the normal (gas passages dry) operating condition.

The reactor was drained and dried (see 3 above) and fuel elements were removed until the loading corresponded to that determined in the wet critical experiment. The procedures used in that experiment were then repeated until criticality was attained.

The dry critical loading was determined to be 59 plate-type elements. The estimated excess reactivity was less than 1% Δ K/K with all control rods fully withdrawn. This loading corresponded to 18.6 kg of U-235; the critical experiment at BMI had predicted that 17.9 kg of U-235 would be required. A typical inverse count rate plot is shown in Figure VI-3.

5. Operating Core Experiment

The purpose of this experiment was to determine that the 61 element core specified in the GCRE-I design (and checked in the full flooded core experiment) would have sufficient excess reactivity for normal operation.

The core loading was increased from the dry critical configuration to the 61 element design loading under an extension of the procedures used in the dry critical experiment. It was determined that the 61 element core had an excess reactivity of approximately 0.7% Δ K/K. This was considered satisfactory for reactor operation.

6. Preliminary Nuclear Instrument Calibration

The purpose of this test was to make a preliminary determination, by irradiation of suitable neutron absorbers in the core, of the

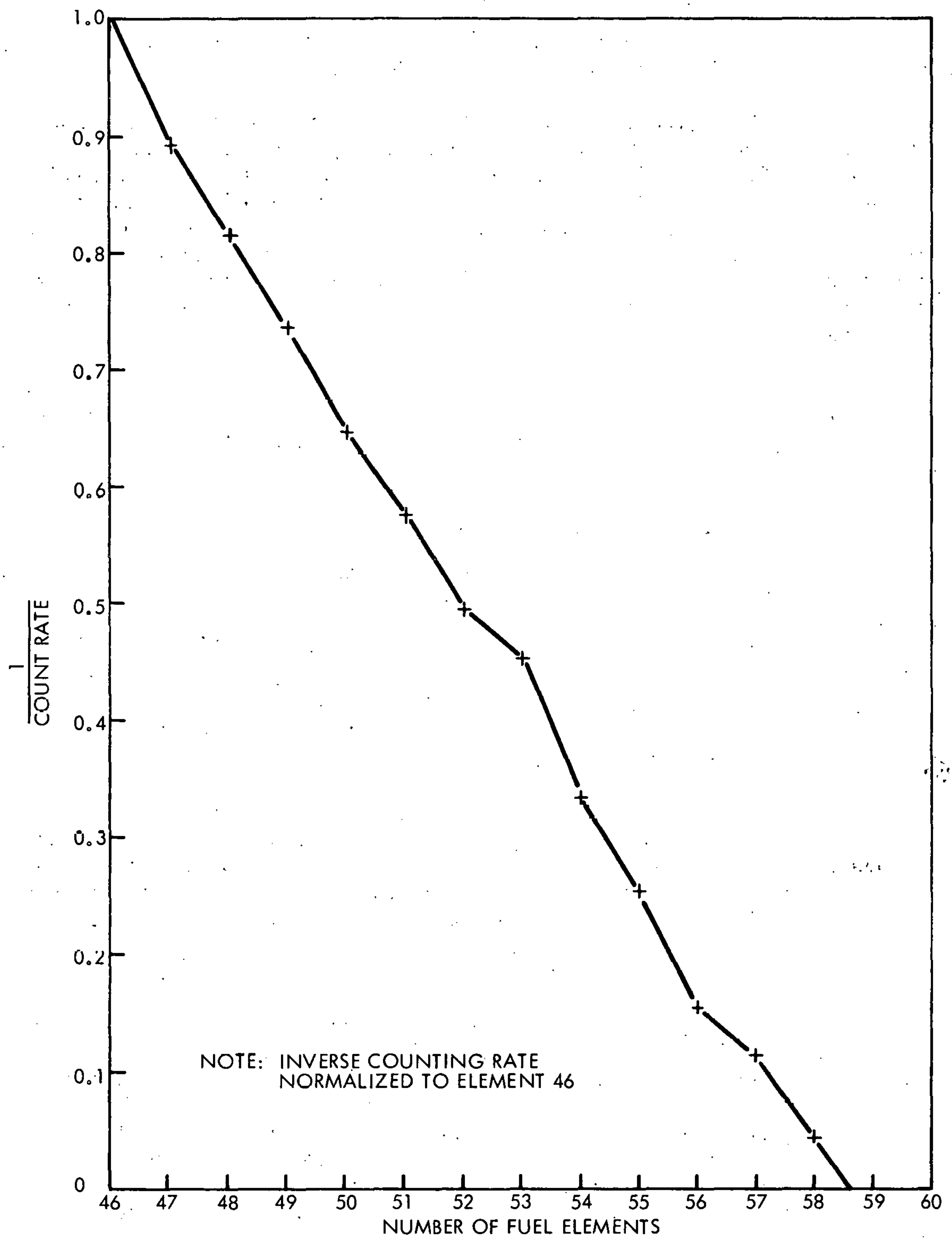

FIGURE VI-3. INVERSE COUNT RATE - GCRE-I DRY CRITICAL EXPERIMENT

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actual operating power of the reactor. This data would provide points of normalization for the nuclear instrumentation read-outs.

Two gold wires and a single U-235 foil were inserted in selected positions in the core and the reactor was operated at a steady, low power for about 40 hr. At the conclusion of this irradiation, the wires and foil were counted and the foil was analyzed chemically. The results of this work permitted a calculation which established, in a preliminary fashion, the actual operating power of the reactor. The calculation indicated that the power was significantly lower than that estimated from the characteristics of the instrumentation, geometry of the ion chambers and reactor, etc.

7. Control Rod Calibration Experiment

The purposes of this experiment were to develop techniques to determine the worth of each individual control rod, to calibrate the fine and shim rods and to determine the worth of individual fuel elements as a function of core position.

The reactor was started up and stabilized at very low power (< 0.01 w) and the control rod configuration was established as follows:

Fine rod - full in

Shim rods - withdrawn equal amounts and controlling the reactor

All other rods - full out

After establishing that the reactor was not on a measurable period, the fine rod was withdrawn a specified amount and the reactor power permitted to rise for at least three decades. The Log-N recorder trace reflected the power rise, and data from this record and the period recorder was subsequently analyzed to estimate the reactivity increase associated with the rod withdrawal. The power rise was terminated by manual actuation of the setback rod.

The stable conditions defined above were re-established and the fine rod withdrawn a greater amount. This procedure was repeated until periods approaching 10 sec were indicated. At that time, the rod configuration was adjusted so that the initial position of the fine rod was 4 in. farther inside the core than the outermost insertion depth attained in the previous series of withdrawals and the shim rods were again balanced at a point farther out of the core than had previously been the case. The series of fine rod withdrawals, period measurements and manual setbacks were repeated until periods approaching 10 sec were indicated. The rods were again repositioned and the test continued in this fashion until period data was attained with the fine rod fully withdrawn.

From the data obtained above, an estimated calibration curve for the fine rod was developed. The setback rod worth was then estimated from an exchange measurement with the fine rod, and the shim rod calibration curves were developed by exchange measurements with the fine and setback rods in combination. In the same fashion, the worth of other rods and of selected fuel elements was estimated by the exchange technique.

The analysis of the data indicated that, with some modification, the technique of positive period measurements to estimate reactivity changes would yield a satisfactory calibration of the fine rod which could then be used in a series of exchange measurements to estimate reactivity worths of other rods and selected fuel elements.

8. Preliminary Moderator Temperature Coefficient Experiment

The purpose of this experiment was to evaluate the procedure for determination of the moderator temperature coefficient of reactivity and accumulate data from which a preliminary estimate of the coefficient could be developed.

The reactor was operated at low power and the moderator (pool) water temperature increased over a period of approximately 40 hr from 68° to 132°F. The critical control rod configuration was observed throughout the period.

Analysis of the data collected indicated that the procedure used was correct and that the coefficient was positive and inversely related to temperature over the range explored. The average value of the coefficient was established at $+ 0.9 \times 10^{-4}\ \Delta K/K\text{-}^{\circ}F$.

9. Flux and Power Mapping Experiment

The purpose of this experiment was to refine the irradiation, counting and calculational techniques used in an earlier experiment (see 6 above) and to develop durther data on core operating power and the flux and power distribution in the core.

Several gold wires and U-235 foils were inserted in the reactor and the unit was operated at low power. Numerous spurious scrams and control rod malfunctions so badly perturbed operations during this irradiation that no conclusive results could be obtained. The counting of wires and foils was completed, however, to provide experience in these operations for later testing.

B. EXPERIMENTS WITH PLATE-TYPE (IZ) FUEL ELEMENTS IN THE STAINLESS STEEL CALANDRIA

The testing described in Section A above was discontinued in mid-April 1960 to permit the replacement of the aluminum calandria with a stainless steel unit and the modification of the control rod actuators. This work was completed early in June and, since the pin-type core loading was not scheduled for delivery until September, it was decided to undertaken a series of tests with plate-type fuel elements. These experiments are summarized below.

1. Initial (Wet) Critical Experiment

The purpose and the procedure of this experiment were the same as those described in Section A-1 above.

Criticality was attained with 52 plate-type fuel elements (16.25 kg U-235). Previous calculations in the BMI critical experiment had indicated that the substitution of a stainless steel calandria for the aluminum unit would require a loading of 38 elements (two more than required in Section A-1 to accommodate the calculated -0.6% Δ K/K change associated with the substitution) to attain criticality. The significant difference between the estimated and actual critical loadings was subsequently attributed to improper assumptions for the neutron age and neutron temperature factors in the calculation. A critical experiment performed at BMI, based on the wet critical data, had indicated that, despite the calculational errors, sufficient reactivity would be available in the dry configuration to permit reactor operation.

2. Full Flooded Core Experiment

The purpose and procedure of this experiment were the same as those described in Section A-2 above.

The core was loaded with 72 plate-type fuel elements; the remaining core position was occupied by the source. In this configuration, it was determined that the reactor would remain subcritical with shutdown rods 2 and 5 and the fine and setback rods withdrawn.

3. Dry Critical Experiment

The purpose and procedure of this experiment were the same as those described in Section A-4 above.

The dry critical loading was determined to be 66 plate-type elements (20.6 kg U-235). This value corresponded exactly with that predicted by the BMI critical experiment discussed in Section A-1 above.

4. Operating Core Experiment

The purpose and procedure of this experiment were identical to those described in Section A-5 above.

The operating core loading was established at 71 plate-type elements.

5. IB-9Ø Flux Mapping Experiment

The purpose of this experiment was to establish the reactivity worth of a pin-type (IB-2L) fuel element in several selected positions in the GCRE-I core. A special test element was fabricated which could be disassembled to permit the removal for counting of several Fe-Mn wires and BeO-UO_2 and UO_2 fuel capsules.

The test element was exposed in five positions in the GCRE-I core by substitution for the plate-type element in the selected location. After irradiation, the wires were cut into 1-in. lengths and counted and axial and intracell flux relationships developed. Absolute values were assigned to the flux data following counting of the fuel capsules.

6. Power Ascension Experiment

The purpose of this experiment was to increase the reactor power in a series of steps to the design level. The procedure specified that the power level be stabilized at several points to permit a complete review of reactor performance data and to develop predictions of performances at the next higher level.

The conduct of the experiment was severely hampered by nuclear instrumentation instability (see discussion in Section IV-D) and,as a consequence, almost one month elapsed between the times of initiation and completion of the test. The maximum power reached was 1.85 Mw(t). Operation was limited at this point by an indicated 1300°F gas outlet temperature from one fuel element. Following the power ascension the reactor operated with no evidence of instability at maximum power for about 48 hr.

7. Photoneutron Experiment

The purpose of this experiment was to predict the magnitude of the neutron source in the GCRE-I reactor resulting from the γ , n reaction with the beryllium in the pin-type core. The experimental procedure involved the operation of the reactor for 48 hr to build up a fission product gamma source followed by the shutdown and flooding of the reactor. The central fuel element was removed from the core and the source inserted in this location. It was planned to compare the shutdown neutron population with the source in the central fuel position with that observed when the source was replaced with a can of BeO. The detectable neutron count rate with the source in the center of the core was too low and, as a consequence, control rods were withdrawn until the reactor was barely subcritical. When the BeO can was substituted for the source and an attempt made to reproduce the earlier control rod configuration, the reactor went critical and neutron counting data could not be obtained. It was decided to abandon this experimental approach in favor of direct measurements to be made after the insertion of the full pin-type core loading.

8. Power Operation

Prior to the startup for an extended power run, two instrumented and one uninstrumented pin-type fuel elements were installed in the plate-type core. The reactor was brought to 1.8 Mw(t) and operated at this power level for 230 hr. During this period, it was determined that the orifice pattern required revision; variations of up to 300°F in the gas outlet temperature from each fuel element were observed (this condition was not surprising inasmuch as the orifice pattern was established for a 61 element core and, when it was apparent that the core size would be 71 elements, the smallest orifices available in the field were installed in the extra 10 peripheral elements). The reactor was shut down and new orifices installed. In the startup following this modification, it was observed that the maximum power attainable was approximately 2 Mw(t) and that the variations in individual fuel element gas temperatures were in the order $\pm$ 30°F. The reactor was shut down early in September for the installation of the pin-type core loading. A total of 750 Mw-hr was accumulated on the plate-type fuel elements.

C. EXPERIMENTS WITH PIN-TYPE (IB-2L) FUEL ELEMENTS IN THE STAINLESS STEEL CALANDRIA

1. August to December 1960

Fifty-seven elements of the pin-type design were delivered to the test site in early August 1960. This fuel element design closely approximated that intended for use in the ML-1. As a consequence, a complete program of neutronic and engineering evaluation was contemplated. The experiments conducted are summarized below.

a. Initial (Wet) Critical Experiment

The purpose and procedure of this experiment were essentially identical with those described in Sections A-1 and B-1 above.

Criticality was attained with a core loading of 45 IB-2L elements (36.4 kg U-235). The BMI critical experiment had predicted that 42 elements would be required.

b. Flooded Full Core Experiment

The purpose and procedure of this experiment were identical with those described in Sections A-2 and B-2 above.

The core loading was increased to 57 elements and it was determined that the reactor was subcritical with shutdown rods 2 and 5 and the fine and setback rods fully withdrawn in the flooded condition.

c. Dry Critical Experiment

The purpose and procedure followed during this experiment were identical with those described in Sections A-4 and B-3 above.

Criticality was attained with a core loading of 53 elements (42.8 kg U-235).

d. Operating Core Experiment

The purpose and procedure of this experiment were identical with those described in Sections A-4 and B-4 above.

It was determined that a loading of 56 fuel elements yielded an excess reactivity in the cold clean condition of approximately 1.6% $\Delta K/K$. This value was considered appropriate assuming that, at nominal power, the xenon buildup would absorb approximately 0.6% $\Delta K/K$.

e. Reactivity Worth Experiment

The purpose of this experiment was to develop a reactivity calibration curve for the fine rod and the reactivity worth of individual fuel elements as a function of position in the reactor core. The fine rod calibration was established by the positive period technique described in Section A-7 above.

The fine rod calibration curve shown in Figure VI-4 indicates a total unshadowed worth of approximately 0.47% $\Delta K/K$. The worths of the setback and shim rods were also measured. This data is presented in Figure VI-5.

The replacement reactivity worth of several selected fuel elements was measured. The rod calibration curves discussed above served as a basis for this data. The positions evaluated and the results are shown in Figure VI-6.

f. Flux Distribution Experiment

The purpose of this experiment was to obtain data from which the flux distribution in the pin-type core could be determined. The flux distribution in a selected fuel element and the effect of control rod shadowing on this distribution were also evaluated.

The experiment was conducted by inserting a special fuel element, containing UO_2 fuel capsules and both cadmium-coated and uncoated copper-manganese wires (see Figure VI-7 and VI-8), in a selected core position and irradiating the element for 15 min at a power level between 150 and 650 w, depending on the core position. Following irradiation, the element was removed and disassembled. The wires and capsules were counted and the flux level and distribution calculated from the counting data.

The axial thermal flux distribution in the core was determined to be very close to that predicted; the absorbing effect of the control rod guides depressed the flux slightly in the center with the result that the flux at the upper and lower ends was somewhat greater than predicted by the PDQ calculations. Typical axial flux and power plots (core position E-5) are shown in Figures VI-9 and VI-10.

The radial thermal flux distribution was slightly higher at the center and lower at the edge than predicted by PDQ calculation. The results of this determination are shown in Figure VI-11. Figure VI-12 shows the radial power distribution.

The power distribution across a selected fuel element position (G-7) and the effect of shim rod shadowing on this distribution are shown in Figure VI-13. As shown in this figure, the total power is reduced and the shape of the distribution curve is altered. Note the significant thermal power depression in the center of the element despite the heavier uranium loading in this region.

g. Moderator Temperature Coefficient Experiment

The purpose of this experiment was to determine the moderator temperature coefficient of reactivity in the IB-2L core over the normal range of operating temperature. The reactor was operated at low power and the critical control rod configuration was observed as the moderator (pool) water temperature was increased by external heating from 68°F to 150°F. The temperature coefficient was calculated from the control rod calibration data.

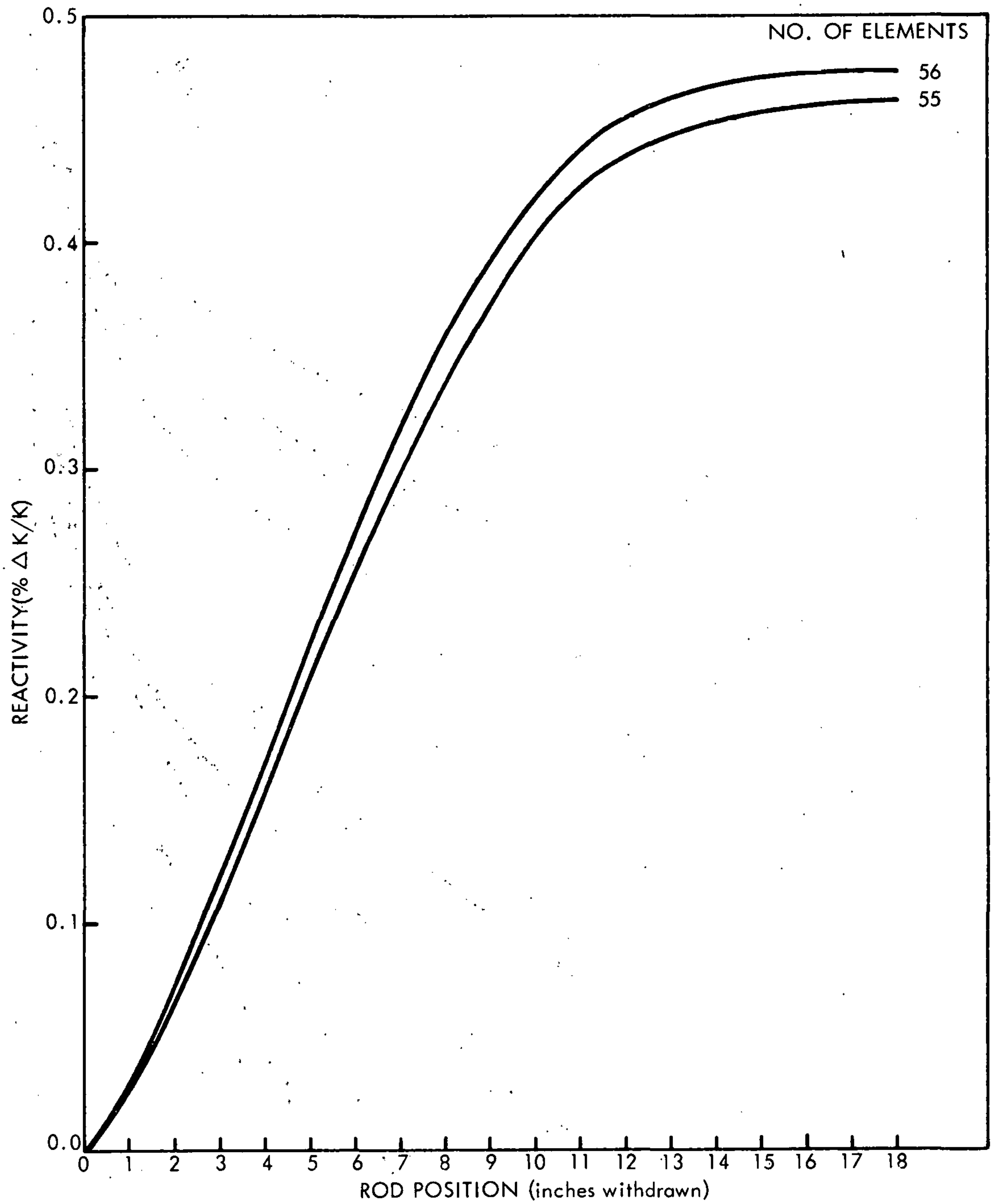

FIGURE VI-4. GCRE-I FINE ROD CALIBRATION CURVE

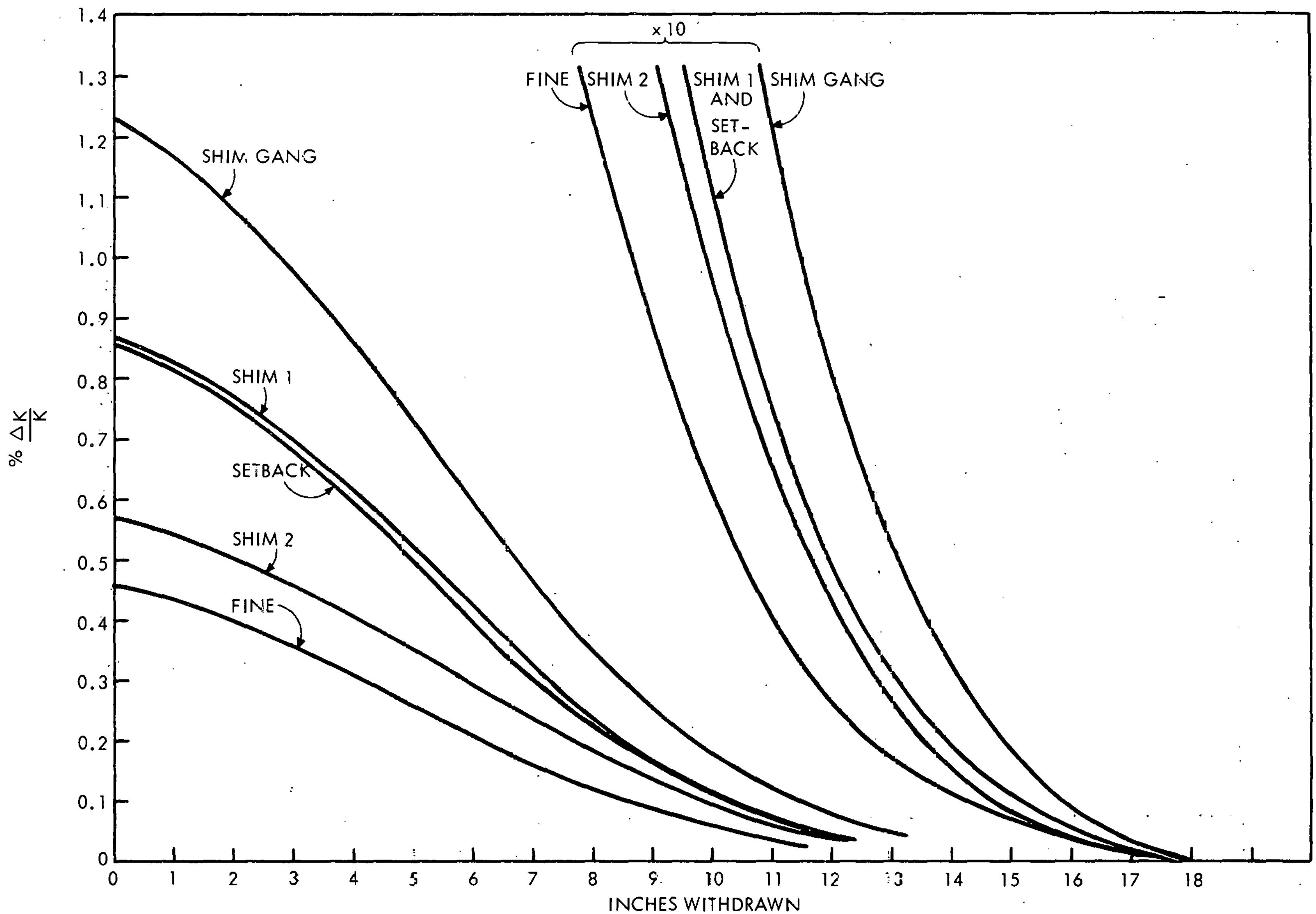

FIGURE VI-5. REACTIVITY WORTH GCRE-I CONTROL RODS

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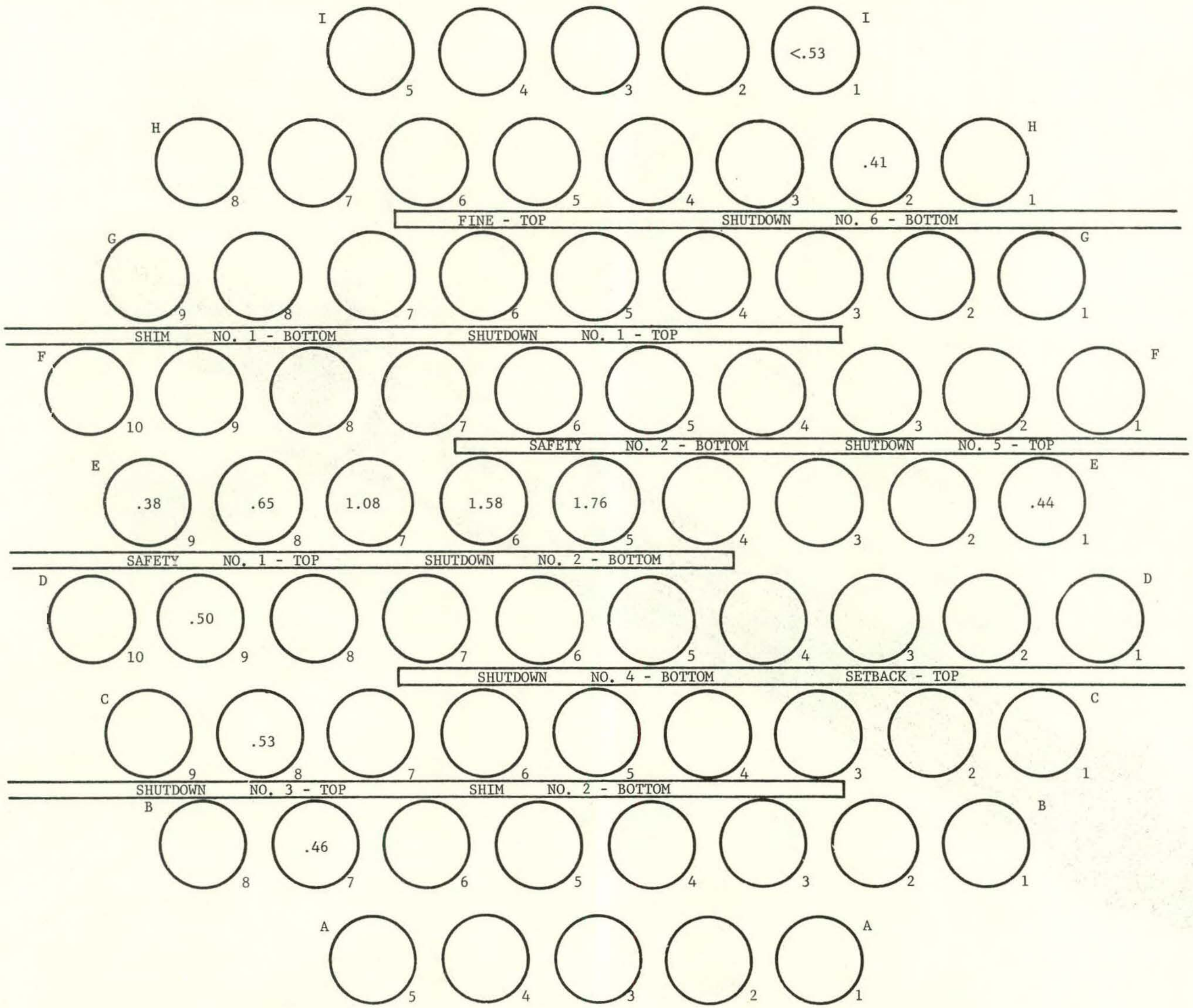

FIGURE VI-6. POSITION WORTH OF GCRE FUEL ELEMENTS
(Values shown are reactivity worths of fuel elements in % Δ K/K)

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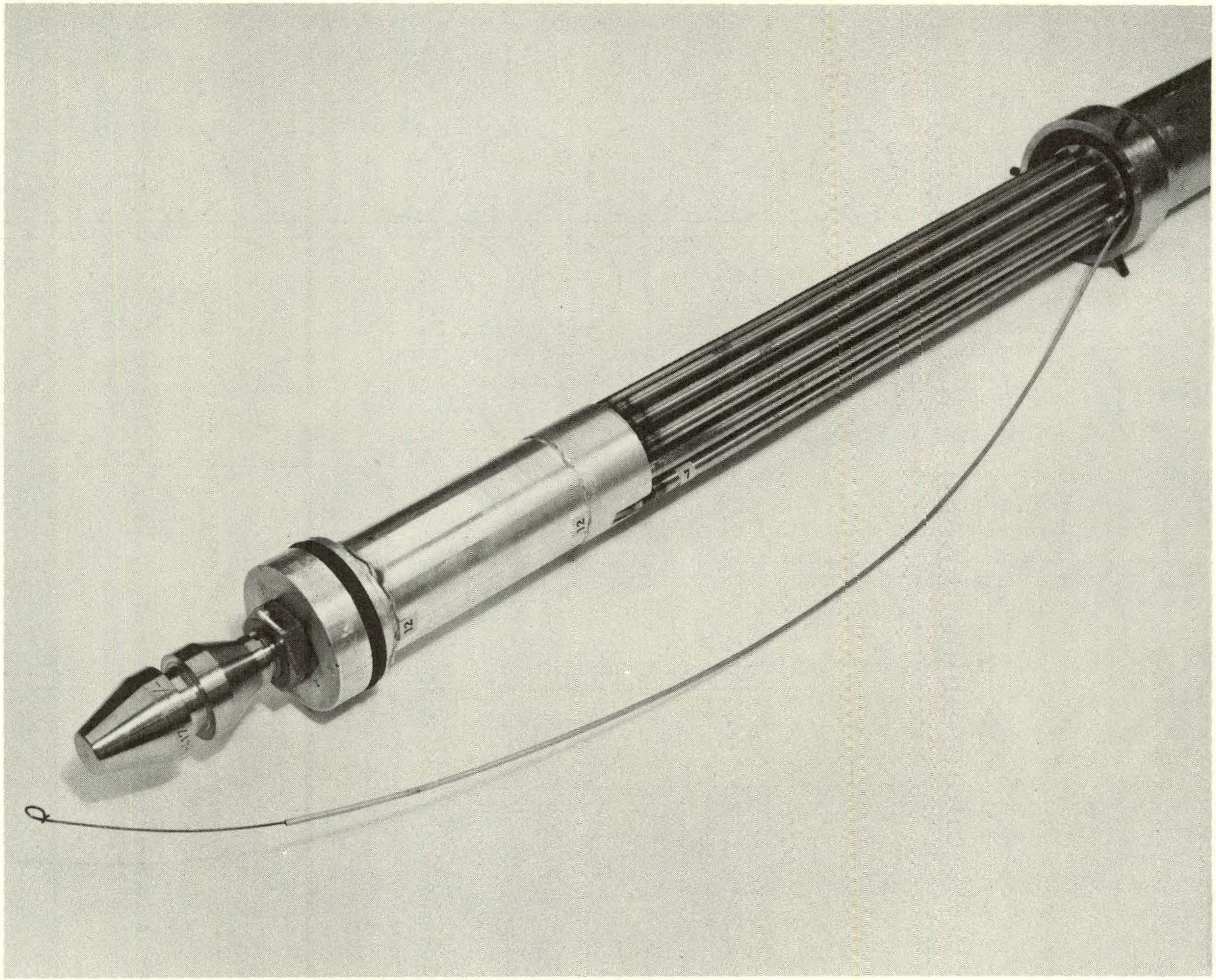

FIGURE VI-7. IB-9Ø R-2 FLUX MEASURING ELEMENT, PARTIALLY DISASSEMBLED

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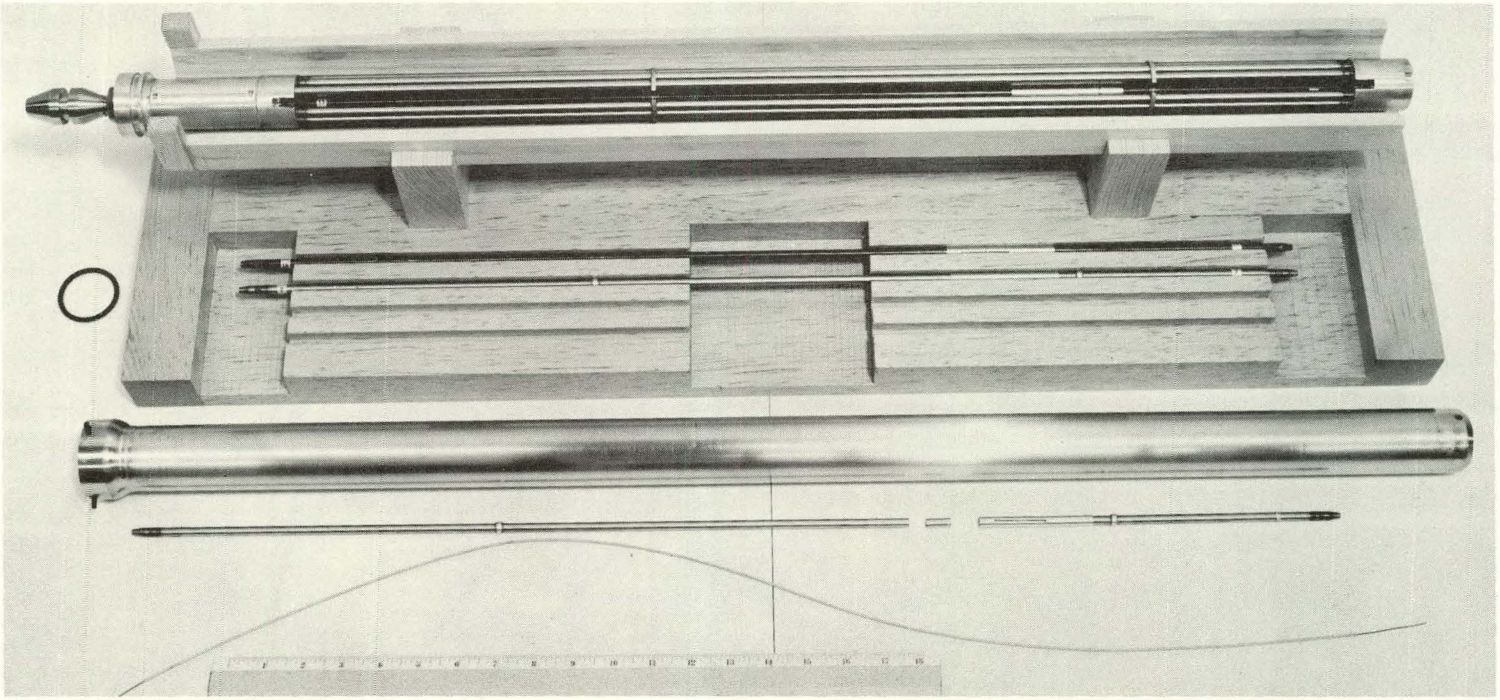

FIGURE VI-8. IB-9Ø R-2 FLUX MEASURING ELEMENT, FULLY DISASSEMBLED, ON ASSEMBLY JIG

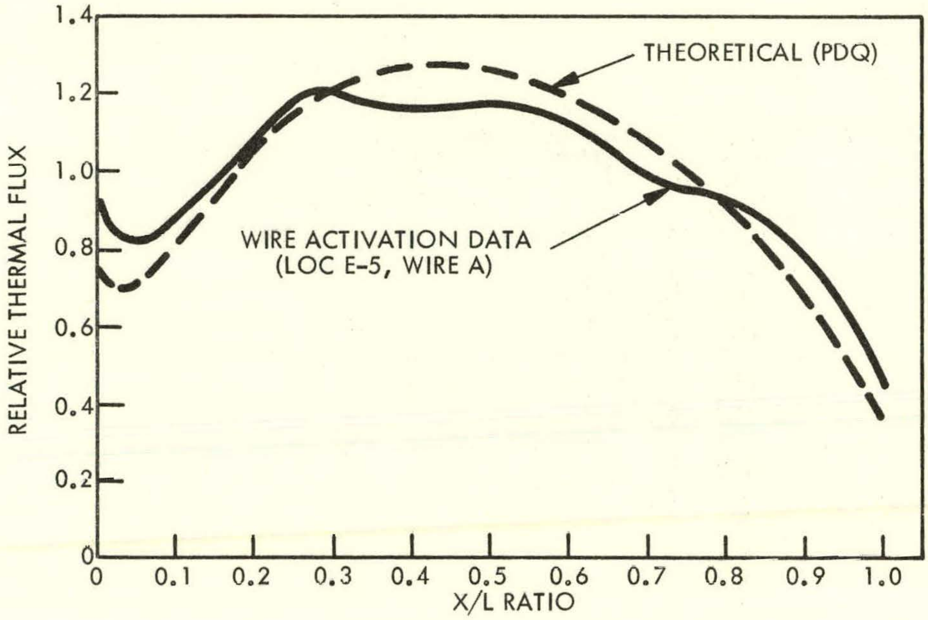

FIGURE VI-9. AXIAL THERMAL FLUX DISTRIBUTION AT CENTER OF GCRE-I

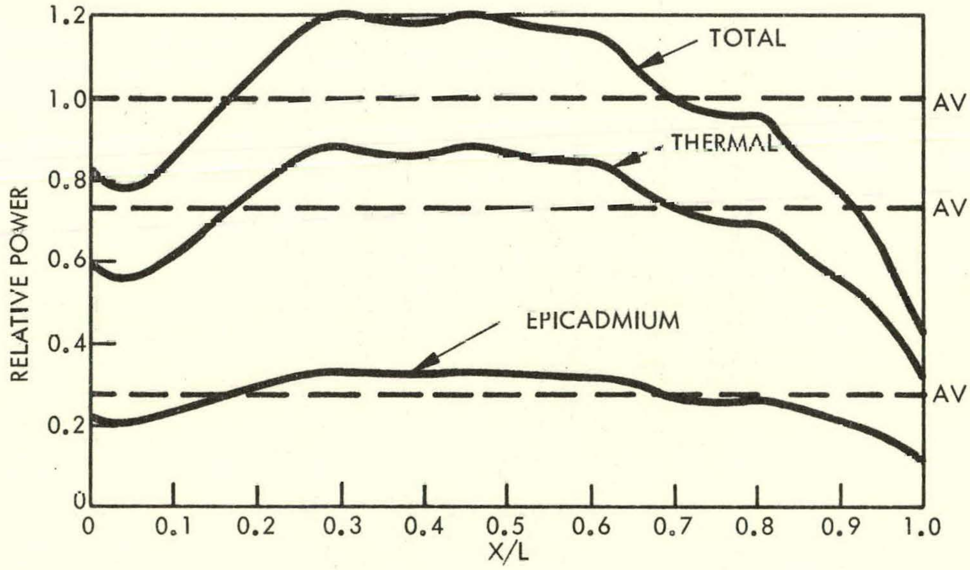

FIGURE VI-10. GCRE-I AXIAL POWER DISTRIBUTION AT ELEMENT POSITION E-5

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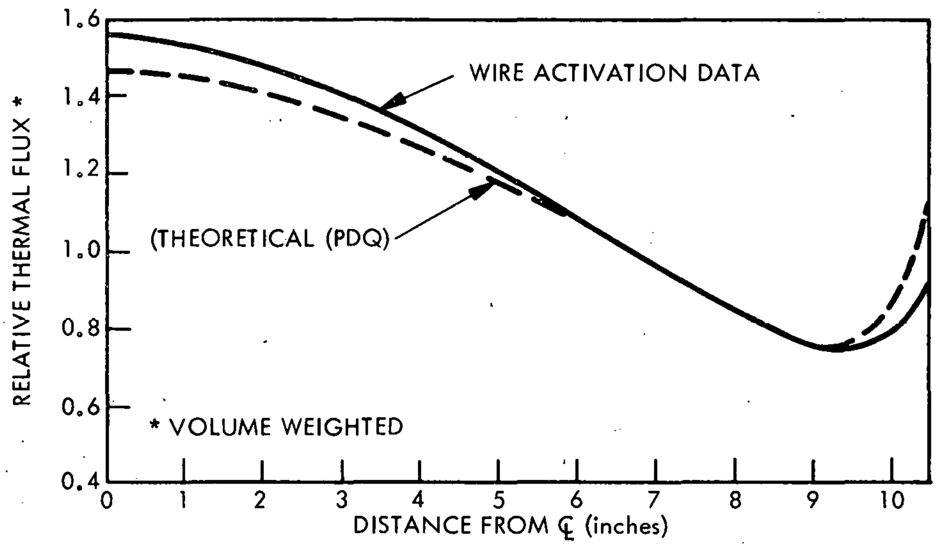

FIGURE VI-11. RADIAL FLUX DISTRIBUTION AT MIDPLANE OF GCRE-I CORE

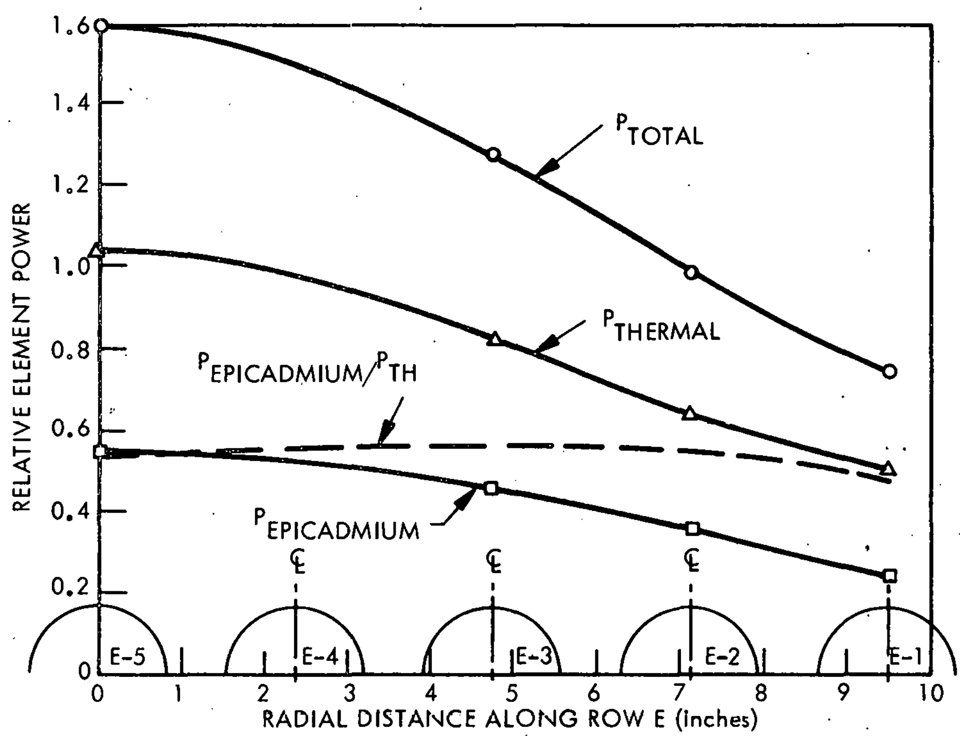

FIGURE VI-12. GCRE-I RADIAL POWER DISTRIBUTION

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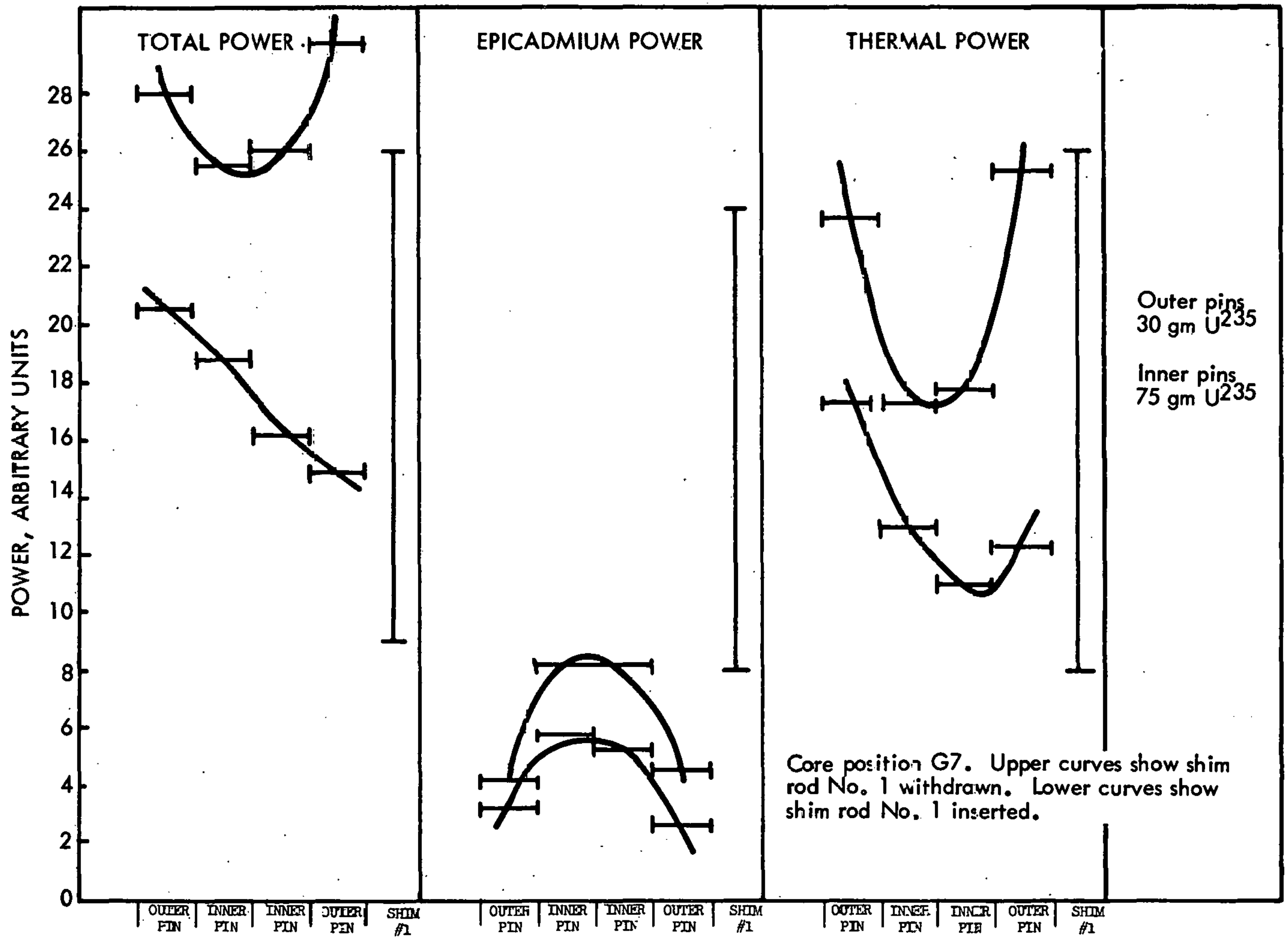

FIGURE VI-13. EFFECT OF SHIM ROD ON PIN-TO-PIN POWER DISTRIBUTION

The coefficient was determined to be + 0.9 x 10^{-4} Δ K/K-^{o}F at about 100^{o}F. The value was more positive at lower temperatures (and less positive at higher temperature); the change was essentially linear over the range of temperature explored.

h. Reactivity Coefficient Experiment

The purpose of this experiment was to develop the data from which the temperature coefficients of reactivity of the fuel and core structure could be determined. The pressure coefficient of reactivity was also evaluated.

With the reactor operating at low power, the temperature of the inlet coolant gas was varied from 900^{o}F to 1150^{o}F by varying the heater outlet temperature. Changes in reactivity were determined from the critical control rod configurations and corrected for the variation in moderator temperature. The coefficient was calculated to be negative and in the order of 10^{-7} Δ K/K-^{o}F.

Variation of the coolant pressure between 50 and 150 psi yielded data which resulted in a calculated pressure coefficient of -7 x 10^{-7} Δ K/K-psi.

The above values were considered preliminary because of mechanical and instrumentation difficulties encountered during the experiment. Plans were immediately made to repeat the test under conditions which would improve the precision of the data and results. In the meantime, an experiment was conducted to evaluate the feasibility of determining the temperature coefficient by oscillation techniques.

In this test, the fine rod was oscillated (alternately inserted and withdrawn) and the resulting reactivity oscillation monitored on Channel IX (the electrometer-actuated linear power nuclear instrumentation channel). This operation was repeated for several different values of rod travel per cycle and a corresponding reactivity variation assigned to the Channel IX trace based on the fine rod calibration data. The temperature of the coolant gas entering the reactor was then oscillated by manipulation of the heater bypass valve and the neutron population in the core monitored by Channel IX. It was planned that the reactivity associated with the temperature oscillation could be determined by comparison of the Channel IX trace with the "calibration" traces established earlier. Although considerable difficulty was experienced in establishing acceptable techniques for the conduct of the experiment and, as a consequence, the precision of the results was open to considerable question, a temperature coefficient of $\pm$ 5 x 10^{-6} Δ K/K-^{o}F at an average temperature of 616^{o}F was determined. An unanticipated phase shift in the system made it impossible to determine whether the coefficient was negative or positive.

In view of the inconclusive results of this test, preparations to repeat the temperature and pressure coefficient experiment were initiated.

i. Power Ascension Experiment

Following the completion of the flange bolt tightening operation (described in Section IV), the GCRE-I reactor with the pin-type core was taken to full power in a series of discrete steps. Loop operating conditions were carefully monitored at each power level and predictions of anticipated conditions at the next higher level were made. Instrumented fuel elements were installed in positions F-5 and F-6 and, except for these elements, all fuel elements were equipped with 0.850 in. diameter orifices.

The maximum power reached was approximately 1.8 Mw(t). Power was limited by the outlet gas temperature of the central seven fuel elements which reached 1250°F. Other pertinent parameters at full power were:

Coolant flow - 18.2 lb/sec

Reactor inlet pressure - 190 psig

Reactor inlet coolant temperature - 800°F

j. Shadowed Fuel Element Temperature Experiment

The purpose of this experiment was to determine the effect on fuel element temperatures of the flux shadowing which resulted from the insertion of a shim rod. The experiment was performed at full power; other system parameters were identical with those described above. The temperatures indicated by the thermocouples on the instrumented fuel elements in positions F-5 and F-6 were observed with shim rod No. 1 completely withdrawn and again with this rod inserted 12 in. No appreciable difference in temperature values was noted. A probable explanation for this anomalous observation was developed based on the fact that the shim rod penetrates the lower (axial) position of the core while, in the IB-2L design, the active portion of the fuel element is positioned in the upper 22 inches of the core. An additional experiment was planned to attempt to develop further information on this subject.

k. Reactor Shutdown Cooling Experiment

The purpose of this experiment was to provide data which would permit a prediction of reactor temperature behavior in the event of a loss of gas coolant flow while the reactor was operating near full power. The experiment was performed in a series of steps which progressively approached approximation of actual reactor conditions during an emergency loss of coolant flow.

Prior to the initiation of the experiment, the reactor was operated at full power for approximately 100 hr to build up the fission product inventory in the reactor fuel. Tests were performed in which the reactor was scrammed and the coolant circulating compressor shut down at varying times thereafter. The temperatures indicated by the thermocouples in the instrumented fuel elements were monitored and correction factors were applied to this data to establish the maximum fuel element temperatures encountered.

In the final run under this experiment with a reactor inlet temperature of 500°F, the reactor was scrammed by stopping the compressor; this resulted in essentially simultaneous reactor scram and cessation of coolant flow. The loop was immediately depressurized and the temperatures observed. The maximum temperature noted was approximately 1100°F. This temperature occurred approximately 30 minutes after scram; beyond this time, the temperature decreased gradually and consistently.

A typical plot of the actual temperatures in the instrumented fuel element (in position F-5) is shown in Figure VI-14. This curve shows the temperatures observed after a reactor scram from normal full power operating conditions followed by compressor shutdown 200 sec later.

The analysis of the data generated during this experiment resulted in three significant conclusions:

1) The capacity of the reactor core to transfer heat to the coolant gas and moderator was greater than had been predicted. The maximum temperatures were observed approximately 30 minutes after the loss of coolant flow and, beyond this time, all temperatures decreased.

2) The reactor heat transfer rate was highly pressure-dependent; the slowest cooling and maximum temperatures occurred when the reactor coolant pressures were lowest.

3) The maximum temperature in the fuel following a loss of coolant flow was significantly less (approximately one-half) of that predicted.

From the analysis, it appeared that natural convective flow of coolant through the reactor was much more significant than had been previously anticipated. Additional experimentation was planned to further evaluate this characteristic of the reactor.

2. <u>December 1960 to April 1961</u>

After completion of the shutdown cooling experiments, the reactor was shut down and orifices in the fuel elements were changed to flatten the radial temperature distribution. Figure VI-15 shows the orifice pattern after this change. The reactor was then operated for about 700 Mw-hr to evaluate the effect of re-orificing and to accumulate experience with the plant during extended operation at steady state conditions. This phase of the program was interrupted on 3 January 1961 when the plant was shut down as a precautionary measure because of an incident at the SL-1 (located about one mile from the GCRE). The GCRE facility served as a communications and decontamination headquarters during the early phases of the investigation of the SL-1 incident. The resumption of GCRE-I operations was delayed by the requirement to decontaminate the facility (contaminated by personnel engaged in SL-1 activities) and to review with the various members of the USAEC the reactor safety and operating procedures in force at GCRE. During this period, a final minor adjustment was made to the fuel element orifice pattern (see Figure VI-16) and all fuel elements were inspected and modified (see Section V-B-5). Operations were resumed on 14 February 1961.

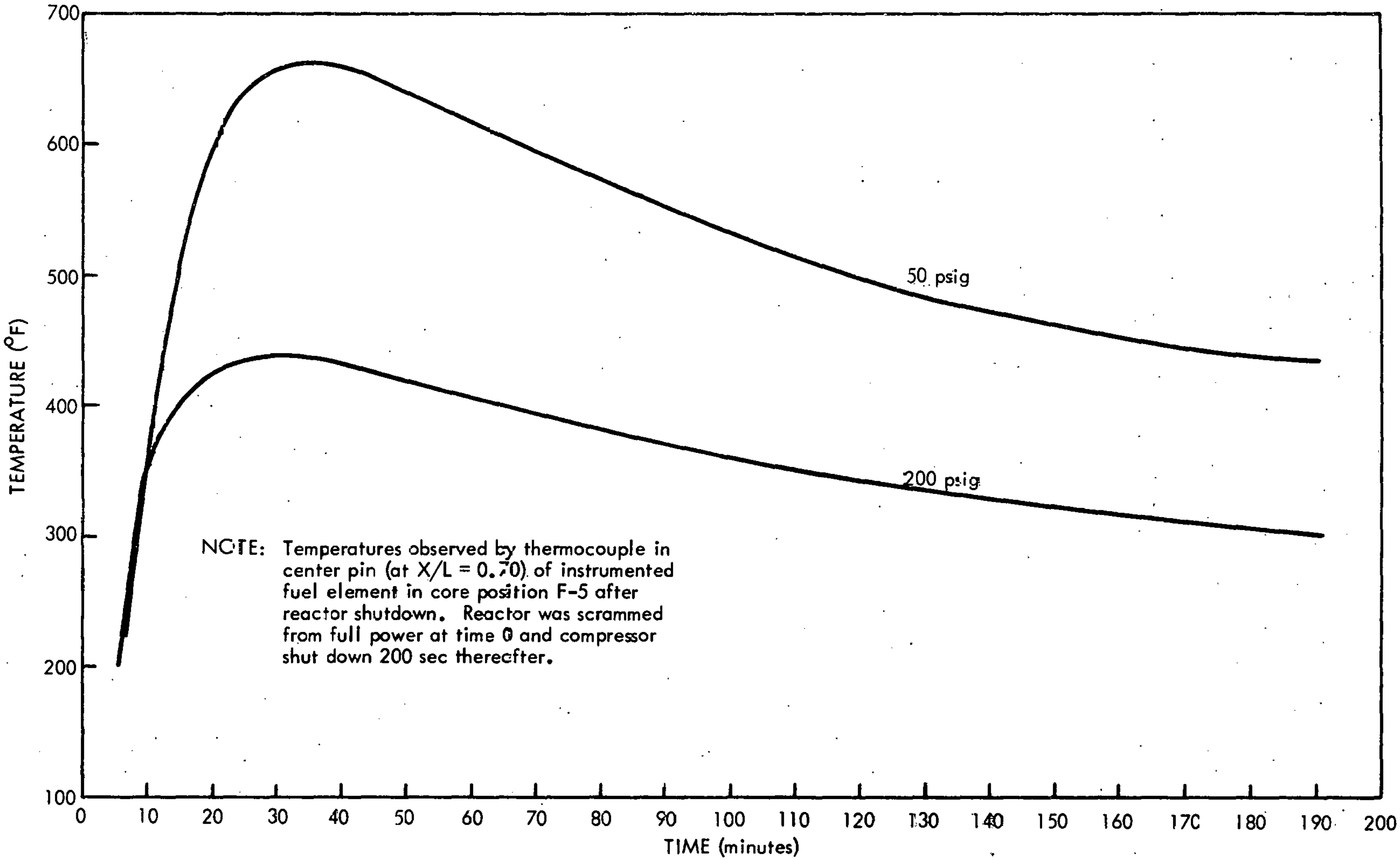

FIGURE VI-14. GCRE-I POST-SHUTDOWN TEMPERATURES

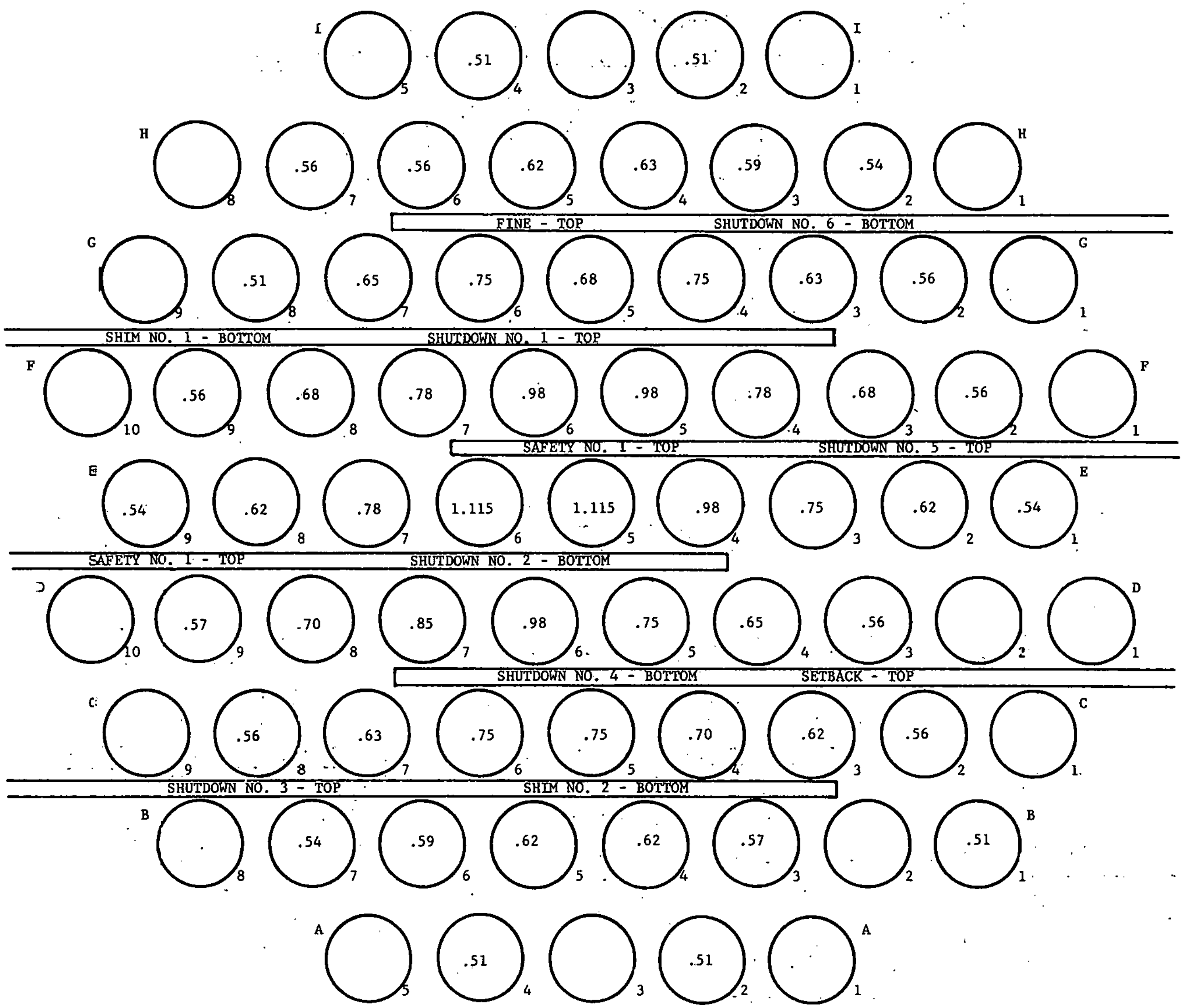

FIGURE VI-15. GCRE-I ORIFICE PATTERN, 6 DECEMBER 1960

(Values shown are orifice diameters in inches)

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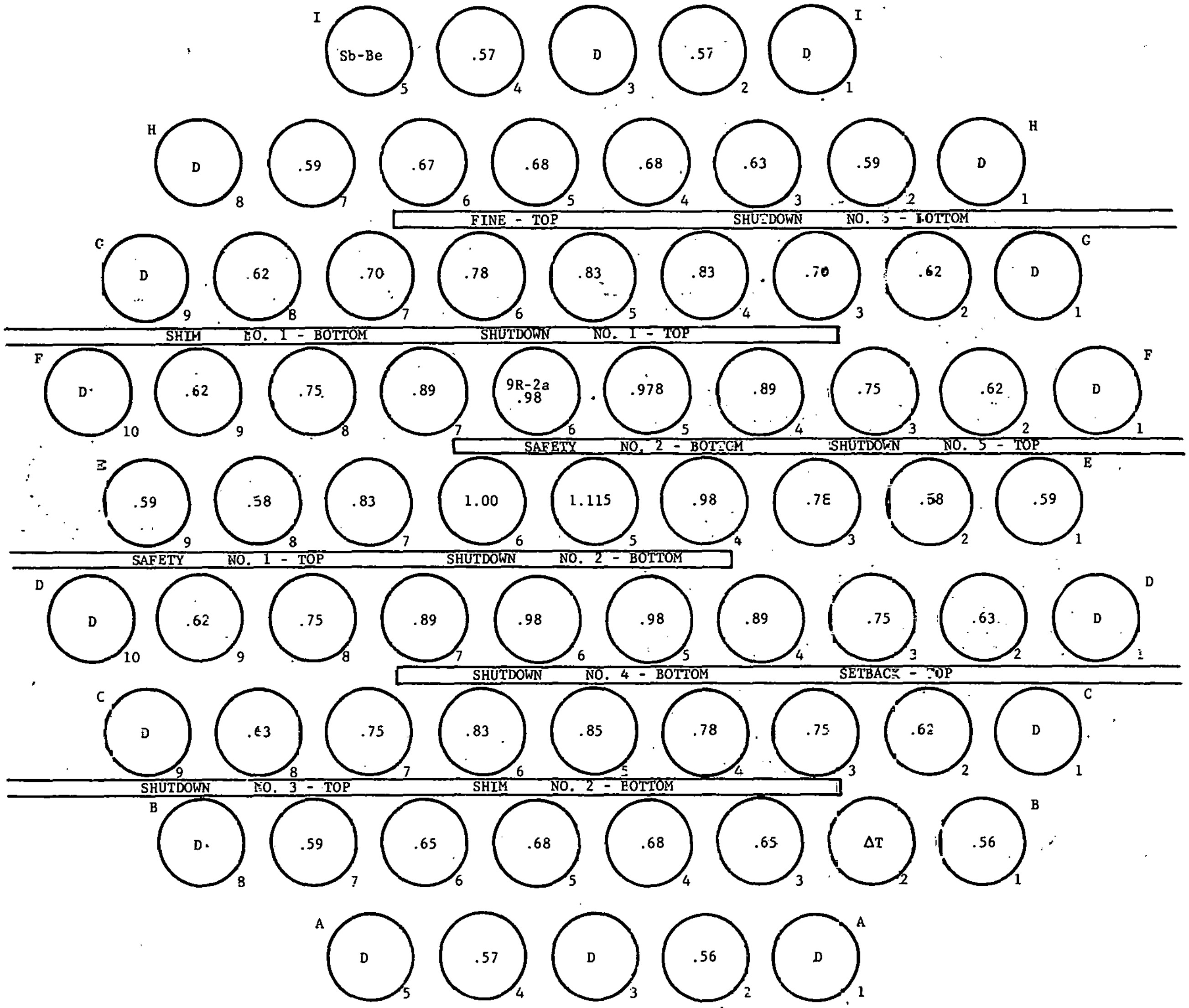

FIGURE VI-16. GCRE-I ORIFICE PATTERN, 24 JANUARY 1961
(Values shown are orifice diameters in inches)

a. Reactivity Coefficient Experiment

The purpose of this experiment was to repeat the work performed earlier (see 1-h above) in an attempt to improve the precision of the values for the core temperature and flow coefficients of reactivity.

The reduction of the data developed during the test yielded coefficients tabulated below. A 95% confidence level was ascribed to these values.

Coolant temperature coefficient $+0.34\ (\pm 0.21) \times 10^{-6}\ \Delta K/K\text{-}^{o}F$

Coolant pressure coefficient $-3.0\ (\pm 1.1) \times 10^{-6}\ \Delta K/K\text{-psi}$

These tests were performed with a 54 element core loading which had an estimated cold clean excess reactivity of about 0.5% Δ K/K. This was equivalent to about 1.0% Δ K/K with the moderator at operating temperature.

b. Photoneutron Experiment

The purpose of this experiment was to measure the power level and decay rate of photoneutrons produced in the BeO in the fuel following a reactor shutdown. This data was essential to the design of the shutdown shield for the ML-1.

The experimental plan involved operation of the reactor at full power for approximately 100 hr following shutdown of the reactor by the insertion of the two shim rods and the setback rod (calculated to provide a reactivity shutdown margin approximately equivalent to that in the ML-1). During the shutdown period, the neutron flux was measured while the moderator water temperature, and the reactor gas coolant inlet coolant temperature, pressure,and flow rate were carefully controlled. The reactor was brought to full power after approximately 24 hr in the shutdown condition and operated for an additional 100 hr. This run was interrupted by several power setbacks and, during the subsequent shutdown period, a malfunction of a shutdown rod rendered much of the data invalid and made the attainment of criticality at the end of the 24 hr period impossible.

The reduction of data taken during the experiment indicated that the ratio of neutron flux level 24 hr after shutdown to that at full power was $7\ (\pm 2) \times 10^{-5}$. Typical decay data are shown in Figure VI-17.

D. MISCELLANEOUS EXPERIMENTS

1. <u>ML-1 Instrumentation Test</u>

The purpose of this test was to check out and calibrate the ML-1 nuclear instrumentation system using the GCRE-I reactor as a neutron source. This approach permitted calibration over the full operating range of the ML-1 instruments.

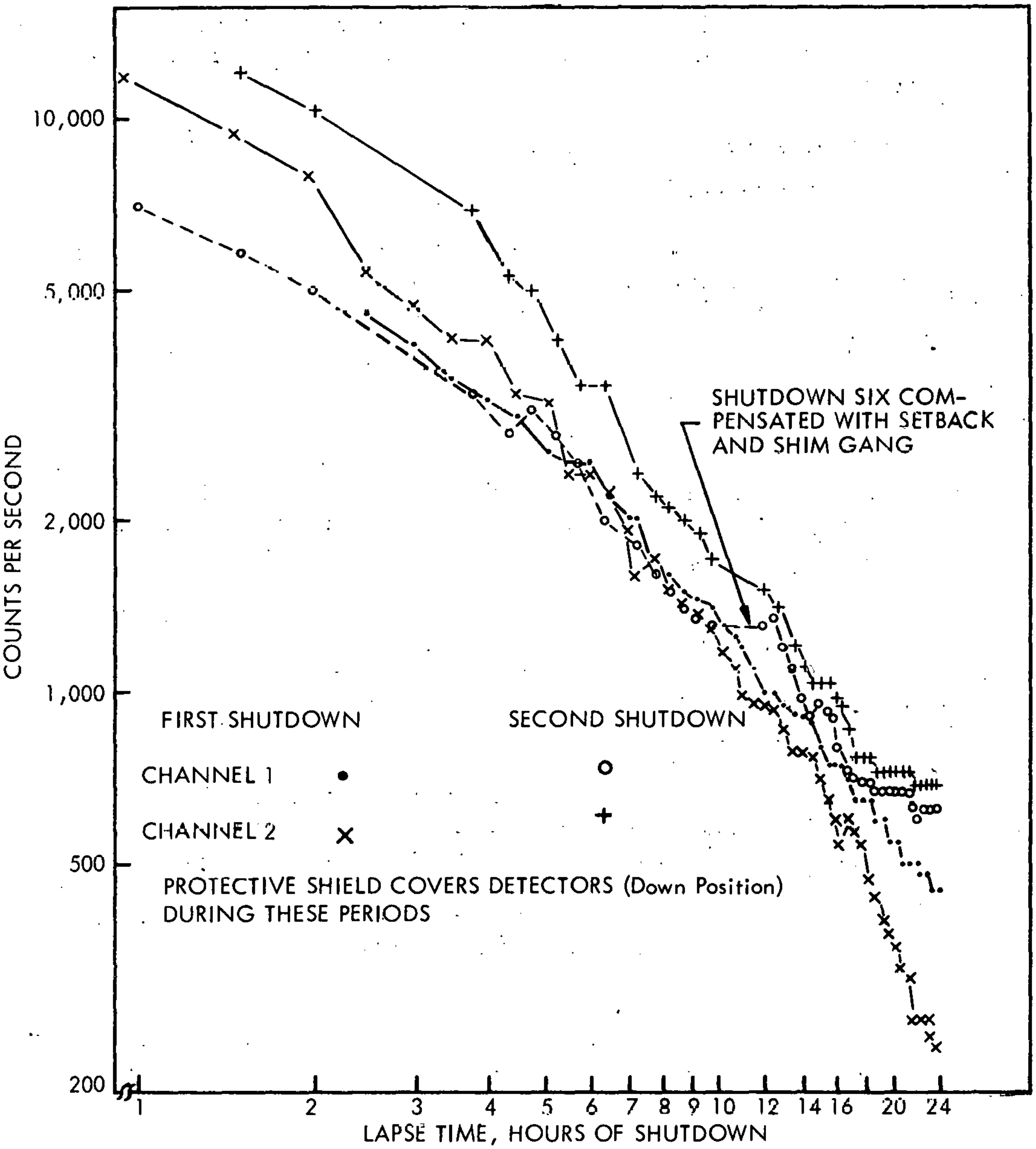

FIGURE VI-17. GCRE-I PHOTONEUTRON EXPERIMENT, SHUTDOWN POWER DECAY DATA

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The ML-1 control cab was positioned inside the GCRE test building and the nuclear instrument detection chambers were installed in thimbles positioned near the GCRE-I reactor. The reactor was started up and the functioning and response of the ML-1 instruments were observed. The sensitivity of the ML-1 system was determined by comparison with the GCRE-I instruments and the discrimination and compensation settings were established in the GCRE-I gamma field.

2. Component Irradiation Program

The purpose of this program was to irradiate small components intended for use in the ML-1 power plant in the gamma and neutron fields of the GCRE-I reactor. A tray was fabricated which could be positioned on top of the GCRE reactor cover during normal reactor operation. The plan was to position the components to be tested in this tray and to observe the effects of radiation of components at periodic intervals.

The first step in the program was the determination of flux levels at the tray. This information was obtained by irradiating detector foils and films placed on the tray during reactor operation. A typical plot of the gamma dose in the tray extrapolated to full power operation is shown in Figure VI-18.

The failure of the calandria and the subsequent deactivation of the GCRE program occurred before any actual irradiation tests could be performed.

3. Power Calculations

A significant effort was expended throughout the GCRE experimental program to develop and refine methods for calculation of the power generated in the reactor. In general, the calculations were performed by use of the formula shown below

$$P = KW \Delta TC_p$$

where

P = total reactor power in kw,

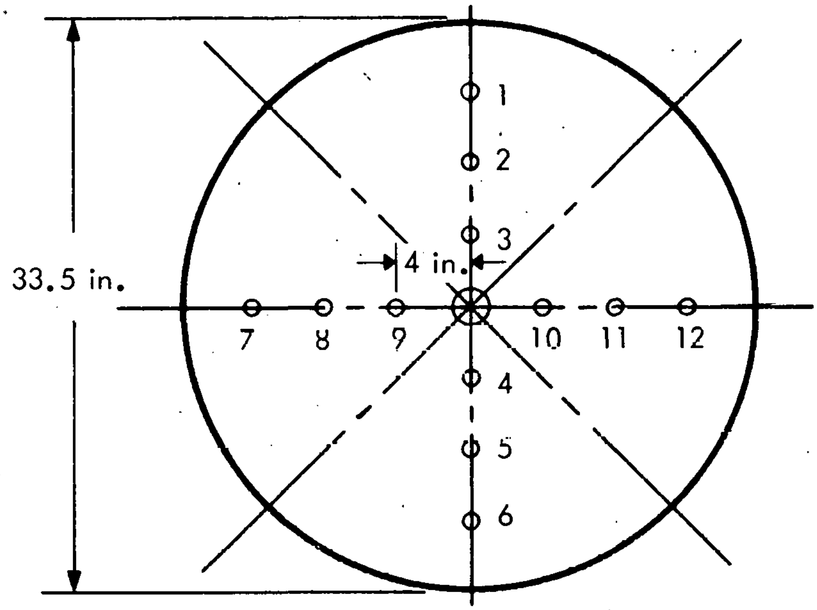
K = a constant (see discussion below),

W = weight rate of coolant flow, lb/hr

ΔT = temperature rise in the gas coolant, $^{\circ}F$, and

C_p = specific heat of the coolant gas, Btu/lb/$^{\circ}F$.

The majority of the work on this subject was associated with defining the proper value of the constant, K. This constant includes the appropriate factor for conversion from Btu/hr to kw and, in addition, includes the factors associated with fission energy deposition in the core structure and moderator as well as the convective heat loss from the core to the moderator. Several calculations were performed to estimate the gamma and neutron energy loss and the final value used in the calculations was 8%. An experiment was performed to estimate the thermal energy loss to the moderator by measuring the temperature drop in the coolant gas with the reactor in the shutdown condition. The final value assigned to the constant was 0.317.



GAMMA FLUX AT 2.2 Mw(t) REACTOR POWER

POSITION	FLUX, $R \times 10^3$
1	20.2
2	27.5
3	31.2
4	34.8
5	28.4
6	23.8
7	25.7
8	28.4
9	34.8
10	34.8
11	31.2
12	23.8

FIGURE VI-18. IRRADIATION TRAY GAMMA DOSES

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Having established the above relationship, correlations were developed to relate the indicated values on the nuclear instrumentation channels with reactor power. A typical plot of this relationship is shown in Figure VI-19.

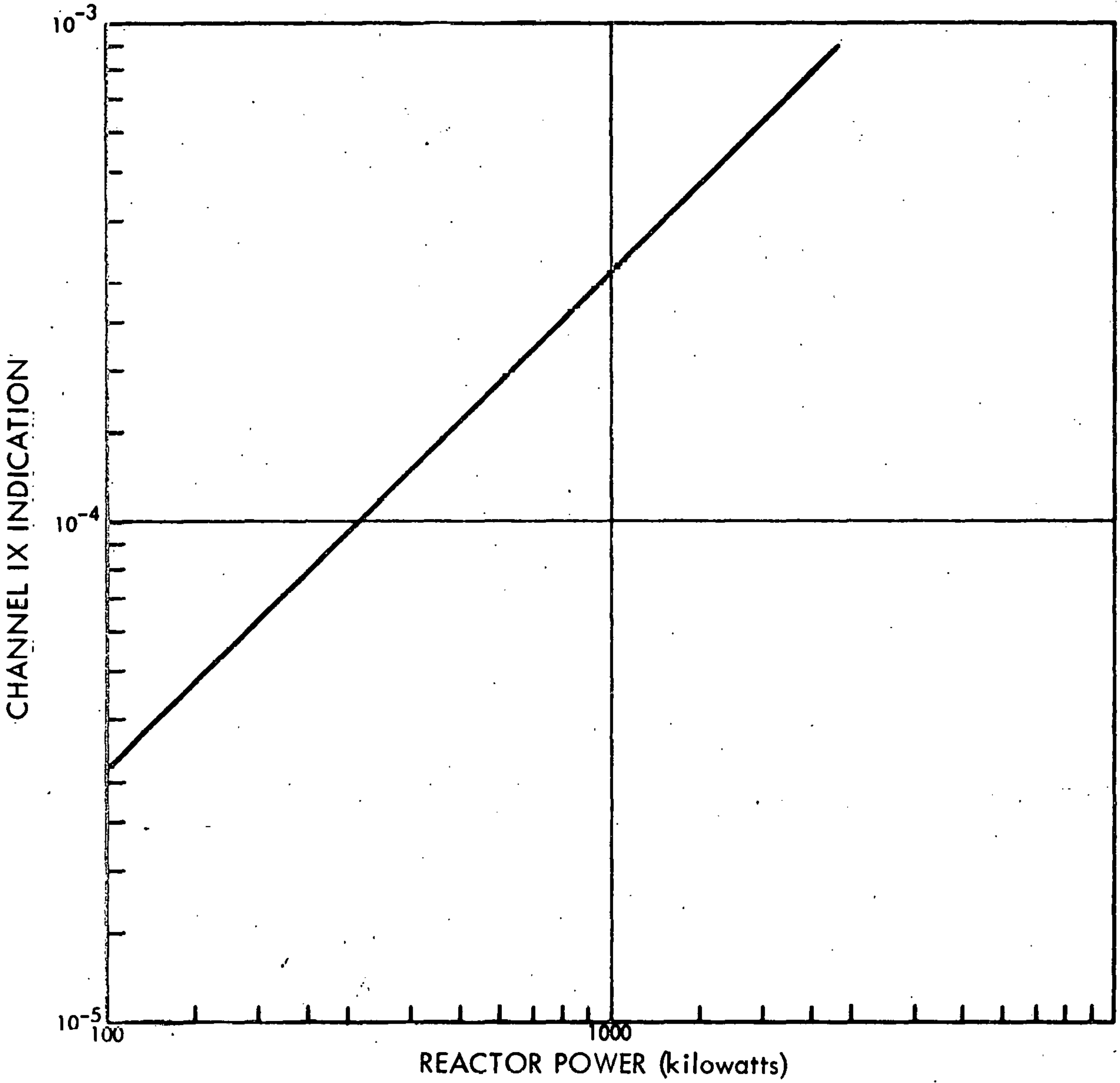

FIGURE VI-19. GCRE-I REACTOR POWER VERSUS CHANNEL IX INDICATION

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VII. OPERATING PHILOSOPHY

All operations at the GCRE were controlled by detailed procedures prepared prior to the initiation of any specific phase of the activity. Such procedures covered the experimental program as well as the "normal" operation of the reactor, the operation of all support equipment in the facility, the maintenance of the reactor and facility equipment and all administrative tasks. The fundamental goal of the procedure system was to ensure, through comprehensive and methodical planning, that no unanticipated event would occur. Strict compliance with the procedures at all levels in the organization was specified as a condition of continued employment.

The development of the procedures began during the period when the design of the facility was in progress. Operations personnel served as liaison agents with the design group and this close association provided the background information required for the initial phases of operational planning and procedure preparation. The first phase of the planning effort involved the development of a set of Technical Standards. These were identified as ANTS (Aerojet-Nucleonics Technical Standards). A typical example of the ANTS is given as Appendix F. The Standards were based on the design data for the reactor and facility equipment and established absolute limits for all operating parameters. For example, the maximum pressures, temperatures, flow rates, etc., for each piece of equipment or subsystem were specified. In addition, the basic rules for operation of the reactor, including the minimum crew size, the assignment of responsibility, etc., were specified.

The next phase of the planning involved the preparation of detailed procedures covering all aspects of the operation, maintenance and administration at the GCRE facility. These procedures were identified as ANSOP (Aerojet-Nucleonics Standard Operating Procedures). Each ANSOP covered a single definable task (for example, pre-startup checkout of the control rod system, startup of the main compressor, reactor startup, normal reactor shutdown, emergency reactor shutdown, dry critical experiment, etc.) and were so comprehensive and detailed that the necessity for independent judgment on the part of the operator was essentially eliminated in all routine and emergency situations. A basic rule in the preparation of ANSOP was that no condition be specified which resulted in violation of the limits presented in the appropriate ANTS.

Prior to final approval and issuance for general usage, all ANTS and all critical ANSOP (those related to reactor operation or involving considerations of nuclear safety) were carefully reviewed and approved by a Procedure Review Board (PRB). The membership of this board included the most experienced and knowledgeable persons at AGN. Special care was taken to ensure that the Board included key design personnel and that all pertinent disciplines were represented. Following approval by the PRB, procedures were submitted by the Supervising Representative to the USAEC for final approval prior to implementation. In selected instances, the procedure or technical standard was also reviewed by the AGN Reactor Safety Committee prior to submission for USAEC approval.

Although the development of ANTS and ANSOP began, as indicated above, during the facility design phase and a major portion of the required procedures were completed prior to the initial operation of the plant, the development of new standards and procedures and the revision to existing procedures continued throughout the operating program. By this mechanism, the ANTS and ANSOP always reflected the experience gained and represented the most current operating conditions.

As a result of the implementation of the policies outlined above, the procedure system accomplished the following desirable objectives:

1) The full capabilities of the most experienced and best trained personnel on the AGN staff were utilized in the conduct of the experimental program.

2) The formulation of detailed procedures and insistence on strict compliance with procedures virtually eliminated the requirement for independent judgment on the part of the operator and thus minimized the possibility for human error and tended to minimize the effects of differences in background and experience among the various members of the operating crew.

3) The preparation of procedures served as an effective training device for the operating crew, inasmuch as such activity required careful study and evaluation of design data and exercise of significant imagination on the part of the writer to integrate the operation under consideration into the overall program and to anticipate non-standard or emergency conditions.

All operating data generated during the GCRE test program was recorded on prepared forms identified as ANSOL (Aerojet Nucleonics Standard Operating Logs). The ANSOL included the instruction log books, the chronological log of operating experience and all routine and special data sheets.

The typical experimental procedure, previously cited as Appendix E, also provides a representative example of the ANSOP. An index of all GCRE ANSOP is presented as Appendix G.

VIII. ORGANIZATION

The organization provided for test operation at the GCRE was staffed in accordance with the following basic considerations:

1) The test site organization should be as small as practical consistent with the ability to perform all normal operations.

2) The organizational philosophy should contemplate the maximum use of all home office resources which could contribute to the effectiveness of the GCRE test program.

3) The organizational philosophy should contemplate utilization to the maximum extent possible of facilities available from the USAEC at NRTS, the NRTS operating contractor (Phillips Petroleum Company) and other operating organizations at the NRTS.

The organization as it existed during most of the program is shown in Figure VIII-1. As indicated in this figure, the group was under the direction of the Operations Manager who reported directly to the AGCRS Program Manager at San Ramon. The responsibility for the operation and maintenance of the facility was delegated to the Operating Superintendent. As indicated, the crew was staffed for continuous 24-hr/day operation. Provisions were made for the conduct of administrative activities under an Administrative Supervisor. Staff specialists in the areas of Health and Safety and Nuclear Engineering were provided.

As shown in the organization chart, a small group existed at San Ramon to assist in the development and evaluation of the experimental program. This group was charged with the responsibility for liaison and coordination with the design organizations resident at San Ramon to assure maximum utilization of the GCRE-I in support of the ML-1 development program.

The training of the GCRE operating crew was initiated more than a year prior to reactor startup. In the early phases of this activity, key personnel were assigned as liaison agents to the organizations responsible for the design and fabrication of the reactor and facility. Informal meetings were held regularly in which the operating personnel reviewed the various aspects of the program with which they were associated. This phase of the training

GCRE

OPERATIONS ORGANIZATION

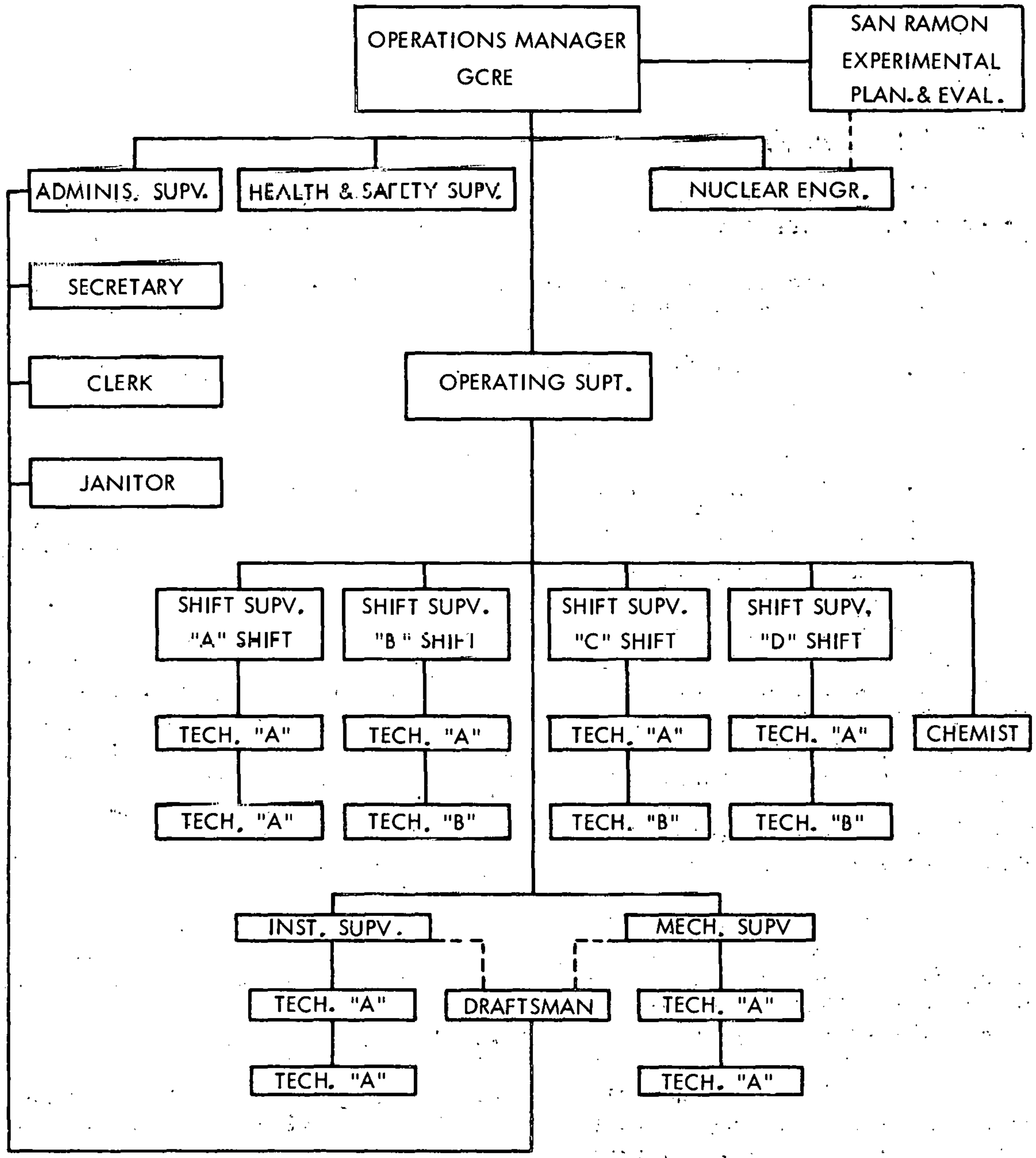

FIGURE VIII-1. GCRE OPERATIONS ORGANIZATION

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resulted in a broad dissemination of detailed knowledge of the design and operation of the equipment.

In the period immediately preceding the relocation of the operating crew to the NRTS test site, a formal training program was conducted at San Ramon. This program consisted of lectures and discussions covering all aspects of the operation. The complete curriculum for this phase of the training is presented in Appendix H. This program was later repeated at the test site for employees who joined the staff at NRTS.

The final and continuing phase of the training program consisted of on-the-job training. This consisted of intensive discussions and briefings prior to the initiation of each experiment and included practice operation, drills and simulated emergency situations to familiarize operating personnel with the equipment and procedures and to evaluate the state of training.

During the early phases of the operation, operating technicians were trained in specific areas of responsibility. This approach assured a high degree of proficiency with a minimum expenditure of time. As the program proceeded and the proficiency of the technicians improved, cross-training was undertaken to provide greater flexibility in the crew and to increase the general level of knowledge of the operators.

REFERENCES

1. The Corps of Engineers, U. S. Army, Army Mobile Power Plant, DC 30.575, New York, Sanderson-Porter Co. (UNC)

2. The Corps of Engineers, U. S. Army, Reactor Survey for Closed-Cycle Gas-Turbine Nuclear Power Plant, DC 31.152, 30 Nov 1955, New York, Sanderson-Porter Co. (CDI)

3. Aerojet-General Nucleonics, Preliminary Design, The Gas-Cooled Reactor Experiment, IDO-28500, San Ramon, Calif.,3 Oct 1957 (SRD)

4. Aerojet-General Nucleonics, The Gas-Cooled Reactor Experiment, Preliminary Evaluation of Beryllium and Beryllium Oxide Heterogeneous Reactors, IDO-28503, San Ramon, Calif.,19 Dec 1957 (SRD)

5. Aerojet-General Nucleonics, The Gas-Cooled Reactor Experiment, Preliminary Design of a Heterogeneous Zirconium Hydride Moderated Reactor System, IDO-28507, San Ramon, Calif.,31 Oct 1958 (SRD)

6. Aerojet-General Nucleonics, The Gas-Cooled Reactor Experiment, Preliminary Design of a Homogeneous Graphite-Moderated Reactor, IDO-28508, San Ramon, Calif., June 1958 (CRD)

7. Aerojet-General Nucleonics, The Gas-Cooled Reactor Experiment, Preliminary Design of a Homogeneous Zirconium Hydride Moderated Reactor, IDO-28509, San Ramon, Calif., Sept 1958 (CRD)

8. Aerojet-General Nucleonics, The Gas-Cooled Reactor Experiment, Preliminary Design of a Homogeneous Beryllium Reactor, IDO-28510, San Ramon, Calif. June 1958 (CRD)

9. Aerojet-General Nucleonics, The Army Gas-Cooled Reactor Systems Program, The GCRE-I Hazards Summary Report, IDO-28506, San Ramon, Calif. Dec 1958 (UNC)

(Continued)

REFERENCES (Continued)

10. Aerojet-General Nucleonics, The Army Gas-Cooled Reactor Systems Program, Tube Bundle Replacement in the GCRE, AGN TM-386, San Ramon, Cali. Sept 1960 (UNC)

11. Aerojet-General Nucleonics, The Army Gas-Cooled Reactor Systems Program, Study of the GCRE Tube Bundle Failure, IDO-28597, San Ramon, Calif. Dec 1962 (UNC)

12. Aerojet-General Nucleonics, The Army Gas-Cooled Reactor Systems Program, The GCRE Control Rod System, AGN TM-387, San Ramon, Calif.,May 1961

APPENDIX A

LIST OF ABBREVIATIONS

AGCRSP	-	Army Gas-Cooled Reactor Systems Program
ANSOL	-	Aerojet Nucleonics Standard Operating Logs
ANSOP	-	Aerojet Nucleonics Standard Operating Procedures
ANTS	-	Aerojet Nucleonics Technical Standards
ARA	-	Army Reactor Area (formerly AREA) at NRTS, Idaho
BMI	-	Battelle Memorial Institute, Columbus, Ohio
GCRE	-	Gas-Cooled Reactor Experiment (In this report, the program or the facility at NRTS.)
GCRE-I	-	Gas-Cooled Reactor Experiment - I (In this report, the test reactor.)
ML-1	-	Prototype gas-cooled nuclear power plant (mobile, low-power, model I).
NRTS	-	National Reactor Testing Station located in southeast Idaho
ORSORT	-	Oak Ridge School of Reactor Technology
PGCR	-	Prototype Gas-Cooled Reactor - the "breadboard" mock-up of the gas-cooled nuclear power plant to demonstrate feasibility
PRB	-	Procedure Review Board

APPENDIX B

ML-1 MILITARY CHARACTERISTICS

This appendix presents the subject characteristics as of 30 November 1960. No revisions have been made since that date.

I. Title: Mobile Nuclear Power Plant, 300-500 kw, ML-1

II. Security Classification:

A. Existence of Project: Unclassified

B. Military Characteristics: Unclassified

C. Plant Design and Performance: Unclassified

III. Proposed Use: There are requirements within the military services for general purpose portable nuclear power plants with electric power outputs in the range from 300 to 500 kw. The ML-1 is the prototype of field units which will be procured to meet these requirements. Examples of military operations requiring a power source without dependence on continuous fuel or cooling water supply are: support of tactical missile systems, air head operations including field hospitals, remote locations, etc.

IV. Environmental Characteristics: The plant shall have the inherent capability for acceptable performance under the Basic Operating Conditions, Extreme Cold Weather Operating Conditions, and Extreme Hot Weather Operating Conditions as established in Paragraphs 7a, 7b, and 7c of AR 705-15; and shall be capable of safe storage and transportation under conditions as established in Paragraph 7d of AR 705-15. A cold weather kit may be utilized for operation under extreme cold weather conditions.

V. Power Characteristics:

A. Nominal Output at "Design Conditions" (Ambient temperature -65°F to 100°F)

(1) Electrical power output in the range of 300-500 kw, at 0.8 Power Factor

(2) 2400/4160 volts, 3 phase, 60 cycles per second

B. Output at "Off Design Conditions"

(1) Net output of the system in the event of hot weather (>100°F) or high altitude operation shall be limited only by the plant heat rejection capacity.

(2) Operation shall be possible up to 500 kw net output except as limited by plant heat rejection capacity.

C. Power Quality

(1) Objective: Steady state and transient frequency and voltage control characteristics as specified by MIL-G-14609 (CE).

(2) Minimum requirement: Steady state and transient frequency and voltage control characteristics as specified by MIL-G-10328A (CE).

D. Other Requirements:

(1) Automatic power level control is required.

(2) The power plant shall operate satisfactorily in parallel with other units of similar rating.

(3) The power plant shall be capable of 50 cycle operation with correspondingly reduced power output.

(4) The power plant shall be treated for the elimination of interference with radio communication in accordance with applicable Signal Corps specifications.

VI. Field Operating Characteristics:

A. The plant unit (connected reactor and power conversion packages) shall be assembled on a standard military semi-trailer (M-172 or M-172A1) and all preparations completed for relocation to an operating site within six (6) hours after unloading from any aircraft.

B. The plant shall be capable of being installed and delivering rated power within twelve (12) hours after arrival at an operating site.

C. The plant shall be capable of relocation to a new site beginning twenty-four (24) hours after reactor shutdown following operation for extended periods at full power. Completion of disassembly and loading shall be within thirty-six (36) hours after reactor shutdown.

D. Nuclear Radiation allowed:

(1) Allowable radiation, twenty-four (24) hours after shutdown at twenty-five (25) ft from the reactor in the direction of the cab (without water shield), is fifteen (15) mr/hr with the power conversion package in place.

(2) No personnel shall receive greater than 3 rem/quarter.

E. Equipment may be operated in any plane within five degrees (5°) from level.

F. Control of the power plant will be from a separate shelter which shall be transportable as a unit on a standard 2½ ton truck (M-35) which, during plant operation, shall be connected to the power plant by quick disconnect electrical cables of sufficient length (up to 500 ft) to permit flexibility in the relative locations of the power plant and control shelter.

G. Plant design shall include necessary safety features for protection of personnel in the immediate area from results of reasonably conceivable mechanical, electrical, or nuclear malfunctions.

H. During plant operation, radiation shielding integral to the plant may be supplemented by field expedient materials (earth, gravel, wood, water) sufficient to reduce radiation levels outside the combined shield to within dosage levels prescribed by the Surgeon General D/A. The integral plant shield (excluding supplemental expedient shielding) shall be adequate to reduce the residual radiation level following full power operation for extended periods of safe levels in time to permit relocation twenty-four (24) hours after reactor shutdown.

I. The operating crew for three-shift operation of the field unit shall consist of no more than seven (7) men (one operator per shift plus mechanic-electrician, radiological safety technician and officer or NCO in charge). The crew will be specially trained for operation, maintenance, and plant installation at the site.

VII. Transportability Characteristics:

A. Objective: Capability of being transported overland (highway and cross-country), by rail, by water, and by aircraft in accordance with AR 705-8, Transportability.

B. Requirements: The ML-1 plant shall be consistent with the following transportation requirements:

(1) General: The "power plant unit" is defined for transportability as the complete integral power plant including reactor, shutdown shielding, power conversion equipment and skid mounting. Control unit, auxiliary power unit, auxiliary equipment packages and trailer are not included in this definition. Required modes of transport are indicated below.

(2) Primary mode of transportation:

(a) The primary mode of transport of the power plant unit shall be overland on standard military semitrailer. Power plant unit integrity is required.

(b) The control unit shall be one package which can be transported intact and which is suitable for both operation and transport on a standard military $2\frac{1}{2}$ ton truck.

(3) Secondary modes of transporation: (power plant unit integrity required).

(a) Secondary modes of transport for the power plant unit where unit integrity is required include:

1. Normal railway freight service U.S. and foreign.

2. Water freight by ship or barge.

(b) In addition, when necessary and where satisfactory clearances can be arranged, the power plant unit shall be transportable trailer-mounted on the above transit types.

(c) Similarly, when truck-mounted, the truck and control unit shall be transportable as one package when necessary and where satisfactory clearances can be arranged for the above transit types.

(4) Secondary modes of transportation: (power plant unit integrity not required).

(a) The power plant unit may be separated into two packages for transport by:

1. USAF C-130.

2. USAF C-124C.

3. Sled.

(b) Trailer mounting is not required for (a) above.

(c) The control unit shall be transportable intact. Truck mounting is not required for (a) above.

(d) The complete power plant including the control unit and auxiliary equipment shall be transportable in one USAF C-133.

(5) Other Requirements:

(a) The power plant shall be transportable in Phase III of airborne operations.

(b) Design for air transport shall conform to requirements of USAF Specification MIL-A-8421A for transportation of material in military cargo aircraft.

(c) Provisions shall be made to permit rapid loading and unloading of each unit by standard weight handling techniques.

(d) Shock and vibration protection shall be provided integral with the plant consistent with each transport type as indicated below:

1. Maximum shock transmitted to cargo bed of transport vehicle without loss of serviceability, utilizing a shock isolation tiedown system:

Direction	Acceleration (g)	Duration of versed sine pulse, (sec)
Fore & Aft	15	0.030
Forward Only	8	0.1
Fore & Aft	3	Static
Vertical Up	4	0.030
Vertical Down	4.5	0.1
Lateral	2	0.030

2. Maximum shock transmitted to cargo bed of transport vehicle without loss of plant serviceability, utilizing rigid tiedown:

Direction	Acceleration (g)	Duration of versed sine pulse, (sec)
Fore & Aft	5	0.1
Fore & Aft	3	Static
Vertical	4.5	0.1
Lateral	1.5	Static

3. Steady vibrations in transit without loss of serviceability.

Source	Peak Amplitude (Inch)	Frequency (cps)
Railroad Flatcar	0.12	20 - 30
Semi-trailer	0.05	>20
Ship	0.05	15 - 20
Aircraft	0.05	>20

(e) Weight:

1. Heaviest package for air transport is 15T.

2. Trailer load (M-172 or M-172A1) is 30T.

3. Control shelter is 2.5T.

(f) Size:

1. Largest package shall be transportable on C-124, C-130, and C-133 aircraft; and on railroad with limits as prescribed by the Berne International Clearance Diagrams.

2. Reactor and power conversion skids shall be transportable as a unit on the M-172 or M-172A1 semi-trailer.

3. Maximum height when trailer-mounted is 150 inches. Power plant height when trailer-mounted may be reduced to 132 inches for the purpose of passing low clearance obstacles outside CONUS.

4. Length and width shall be compatible with a standard military semi-trailer (M-172 or M-172A1).

(g) When trailer-mounted and in transit, the plant shall have the capability of shallow fording in fresh or salt water, as defined in AR 705-2300-8. Extreme caution must be taken to avoid a nuclear accident due to core flooding.

VIII. Logistical Characteristics:

A. The plant shall operate with no more than periodic field maintenance service for 10,000 hr between major overhaul.

B. Routine plant maintenance shall be performed every 30 days or at greater intervals, as determined by operational testing.

C. The plant shall have 50,000 hr total life.

D. The plant shall be capable of full power operation for 10,000 hours between refueling operations. Refueling may be accomplished in the field.

E. The plant shall be capable of operation without a continuous water supply. Any initial quantity of water (for shielding or reactor moderator) will be kept to a minimum and of a quality which can be supplied by standard military purification equipment capable of delivering 600 gph.

F. Radioactive waste from the plant will be kept to a minimum. Disposal of radioactive waste will be in accordance with field procedures approved by the Chief Chemical Officer. (NOTE: Disposal will be subject to AEC approved procedures at NRTS.)

G. The plant will be completely self-sufficient to permit steady state operation without the need for auxiliary equipment, except for process fluid storage and makeup. For startup and shutdown operations, the plant should require no more than a 45 kw diesel generator and necessary construction equipment for erection of the supplemental expedient shield. Equipment for expedient shield construction shall not be part of the overall plant equipment.

H. The power plant shall be equipped with all special purpose tools required for normal maintenance and startup, but not for refueling. Special equipment and procedures required for field refueling shall be specified. The prototype plant shall be equipped with a Maintenance Package as referenced in AR 705-5 and AR 750-6. The field plant shall be equipped with complete operating manuals defining detailed operating, maintenance, installation and relocation procedures. The manuals shall conform to the Publication Standards and Style Guide for Corps of Engineers Equipment Publications.

APPENDIX C

BIBLIOGRAPHY OF GCRE-I REPORTS

This appendix presents the bibliography of formal reports related to the GCRE program. The many informal notes and limited distribution reports supporting the program work are not listed. A brief abstract of the report material is included where classification rules will permit and where the subject matter is not obvious from the title.

The corporate author of all reports listed was Aerojet-General Nucleonics, and the place of publication in all instances was San Ramon, California.

Report	Classification
Gas-Cooled Reactor Experiment, Preliminary Design, IDO-28500, 3 Oct 1957 (This report covers work done to 30 June 1957)	SRD
Gas-Cooled Reactor Experiment, Preliminary Evaluation of Beryllium and Beryllium Oxide Heterogeneous Reactors, IDO-28503, 19 Dec 1957	SRD
Army Gas-Cooled Reactor Systems Program, Preliminary Design of Heterogeneous, Zirconium-Hydride Moderated Reactor System, IDO-28507, 31 Oct 1958	CRD
Gas-Cooled Reactor Experiment, Preliminary Design of a Homogeneous, Graphite-Moderated Reactor, IDO-28508, Jan 1958	CRD
Gas-Cooled Reactor Experiment, Preliminary Design of a Homogeneous Zirconium Hydride-Moderated Reactor, IDO-28509, Sept 1958	CRD
Gas-Cooled Reactor Experiment, Preliminary Design of a Homogeneous Beryllium Reactor, IDO-28510, Jun 1958	CRD

Army Gas-Cooled Reactor Systems Program, Reactor Systems Neutronics, IDO-28511, 7 Nov 1958	CRD
Army Gas-Cooled Reactor Systems Program, GCRE-Heat Transfer Considerations, IDO-28512, 24 Nov 1958	CRD
Preliminary Report of Health and Safety Aspects AGN TM-300, 3 Jul 1957	SECRET
Preliminary Evaluation of Beryllium and Beryllium Oxide/Heterogeneous Reactors, AGN TM-304, 20 Jul 1957	SECRET
Army Gas-Cooled Reactor Systems Program GCRE-I Hazards Summary Report, IDO-28506, Dec 1958	UNC
AGCRSP, GCRE-I Hazards Summary Report, Addendum I, IDO-28506, Mar 1959	UNC
AGCRSP, GCRE-I Hazards Summary Report, Addendum II, IDO-28506, Feb 1960	UNC
AGCRSP, GCRE-I Hazards Summary Report, Addendum III, IDO-28506, May 1960	UNC
Study of the GCRE Tube Bundle Failure, IDO-28597, Dec 1962 (History of fabrication, testing and operation of the tube bundle is reviewed, as well as the post-failure examination and metallurgical investigations. It is concluded that the failures occurred as the result of loss of coolant flow and subsequent boiling, which initiated complex corrosion processes in highly stressed areas of the pressure tubes in the region of the upper tube sheet coolant passages.)	UNC
Proposal for GCRE Core Test, AGN TM-302, 19 Jul 1957	C
Facility Requirements and Experimental Program for GCRE, AGN TM-303, 25 Jul 1957	S
Fission Product Monitoring System Experimental Mixer for Sampling Section, AGN TM-306, 23 Aug 1957 (A proposal to design a mixer for the gas sampling system which will enable the sampling tube to collect fission products representative of the size of the break in the fuel element, regardless of the location of the break within the fuel element assembly.)	UNC
Reactor Control System for the GCRE, AGN TM-307, Undated	CONF
Summary of Experiments with Gas Sampling Mixers, AGN TM-312, 16 Dec 1957	CONF

Army Gas-Cooled Reactor Technical Steering Committee GCRE Progress Report, AGN TM-320, 28 Mar 1958	C
Evaluation Test - Three-Inch Keystone Valve With a Hypalon Seat, AGN TM-326, 2 Jun 1958 (This valve was contemplated for use in the GCRE reactor secondary loops. The valve was deemed suitable for use under certain conditions.)	UNC
Engineering Core Test, AGN TM-344, Undated	CRD
Utilization of the GCRE in Support of the ML-1A Program, AGN TM-376, 5 May 1960 (Proposes a program for maximum utilization of the GCRE-I facility in providing experimental data for the ML-1 during the period 1960 through June 1962.)	UNC
The Thermoelastic Analysis of the GCRE-I Tube Sheet and Plenum Chamber, AGN TM-377, 17 May 1960 (Presents the analytical development of the stress distribution in a perforated tube sheet and a cylindrical plenum chamber which are subjected to high temperatures and pressures.)	UNC
Tube Bundle Replacement in the Gas-Cooled Reactor Experiment, AGN TM-386, 27 Jul 1960 (Describes the design and testing of the aluminum tube bundle as well as operating experience of the aluminum tube bundle. Also describes the design, fabrication and installation of the stainless steel tube bundle.)	UNC
The GCRE Control Rod System: Operational Difficulties and Design Modifications, AGN TM-387, 19 Aug 1960 (Describes the design modifications, resulting from testing and operational experience, of the control rod system in the GCRE to permit reliable, stable, maintenance-free operation of the rods.)	UNC
GCRE Tube Bundle Failure, AGN TM-394, Mar 1962 (Describes the original stainless steel tube bundle, summarizes the operational experience and testing data up to the time of failure and discusses the mechanism of failure.)	UNC
Gas-Cooled Reactor Experiment Progress Report, 21 Apr - 20 May 1957, IDO-28501, 23 Sept 1957	CRD
-------21 May-20 June 1957, IDO-28302, 23 Sept 1957	CRD
-------21 June-20 July 1957, IDO-28504, 7 Jan 1958	CRD

GCRE Semiannual Progress Report, 1 November 1956 to 30 June 1957, IDO-28505, 20 Feb 1958	CRD
GCRE Progress Report, 21 July - 31 Aug 1957, IDO-28515 7 Jan 1958	CRD
------- 1 September to 30 September 1957, IDO-28516, 15 Jan 1958	CRD
GCRE Monthly Progress Report, October 1957, IDO-28517, 18 Feb 1958	CRD
------- November 1957, IDO-28518, 8 Jan 1958	CRD
GCRE Semiannual Progress Report, 1 July - 31 December 1957, IDO-28519, 26 May 1958	CRD
GCRE Monthly Progress Report, January 1958, IDO-28520 28 Feb 1958	CRD
------- February 1958, IDO-28521, 20 Mar 1958	CRD
------- March 1958, IDO-28523, 28 Apr 1958	CRD
------- April 1958, IDO-28524, 28 May 1958	CRD
------- May 1958, IDO-28525, 20 Jun 1958	CRD
GCRE Semiannual Progress Report, 1 January - 30 June 1958, IDO-28526, 12 Oct 1958	CRD
GCRE Monthly Progress Report, July 1958, IDO-28527 29 Aug 1958	CRD
Gas-Cooled Reactor Systems Program Monthly Progress Report, August 1958, IDO-28528, 30 Sept 1958	CRD
Army Gas-Cooled Reactor Systems Program Monthly Progress Report, September 1958, IDO-28529, 31 Oct 1958	CRD
------- October 1958, IDO-28531, 24 Nov 1958	CRD
------- November 1958, IDO-28532, 30 Dec 1958	CRD
AGCRSP Semiannual Progress Report, 1 July-31 December 1958, IDO-28533, 28 Feb 1959	CRD
AGCRSP Monthly Progress Report, January 1959, IDO-28535, 28 Feb 1959	CRD
------- February 1959, IDO-28536, 31 Mar 1959	UNC

------- March 1959, IDO-28538, 30 Apr 1959	UNC
------- April 1959, IDO-28540, 5 May 1959	UNC
------- May 1959, IDO-28541, Jun 1959	UNC
AGCRSP Semiannual Progress Report, 1 January - 30 June 1959, IDO-28592, Jul 1959	UNC
AGCRSP Monthly Progress Report, July 1959, IDO-28543, Aug 1959	UNC
------- August 1959, IDO-28544, 4 Sept 1959	UNC
------- September 1959, IDO-28545, 1 Oct 1959	UNC
------- October 1959, IDO-28546, 1 Nov 1959	UNC
------- November 1959, IDO-28548, 10 Dec 1959	UNC
AGCRSP Semiannual Progress Report, 1 July - 31 December 1959, IDO-28549, Jan 1960	UNC
AGCRSP Monthly Progress Report, January 1960, IDO-28551, 16 Feb 1960	UNC
------- February 1960, IDO-28553, 8 Mar 1960	UNC
------- March 1960, IDO-28554, 23 Apr 1960	UNC
------- April 1960, IDO-28556, 17 May 1960	UNC
------- May 1960, IDO-28557, 8 June 1960	UNC
AGCRSP Semiannual Progress Report, 1 January - 30 June 1960, IDO-28558, July 1960	UNC
AGCRSP Monthly Progress Report, July 1960, IDO-28559, 5 August 1960	UNC
------- August 1960, IDO-28562, 7 Sept 1960	UNC
------- September 1960, IDO-28563, 7 Oct 1960	UNC
------- October 1960, IDO-28565, Nov 1960	UNC
------- November 1960, IDO-28560, Dec 1960	UNC
AGCRSP Semiannual Progress Report, 1 July - 31 December 1960, IDO-28567, Jan 1961	UNC
AGCRSP Monthly Progress Report, January 1961, IDO-28568, 16 Feb 1961	UNC

-------February 1961, IDO-28569, 6 Mar 1961	UNC
-------March 1961, IDO-28570, 11 Apr 1961	UNC
-------April 1961, IDO-28571, 10 May 1961	UNC
-------May 1961, IDO-28572, 10 Jun 1961	UNC
AGCRSP Semiannual Progress Report, 1 January - 30 June 1961, IDO-28573, Jul 1961	UNC

APPENDIX D

GCRE SYSTEM PROCESS FLOW DIAGRAMS

This appendix presents schematic flow and instrumentation diagrams for the major GCRE process systems.

The systems are identified by the "shorthand" designations developed by the design and operating personnel. These are defined below:

GNL - Main nitrogen loop

GN - Waste and make-up nitrogen

WD - Demineralized water

WP - Pool water

WA - Liquid waste

WC - Cooling water

AI - Instrument air

AU - Utility air

MAIN COOLANT LOOP - EQUIPMENT LEGEND

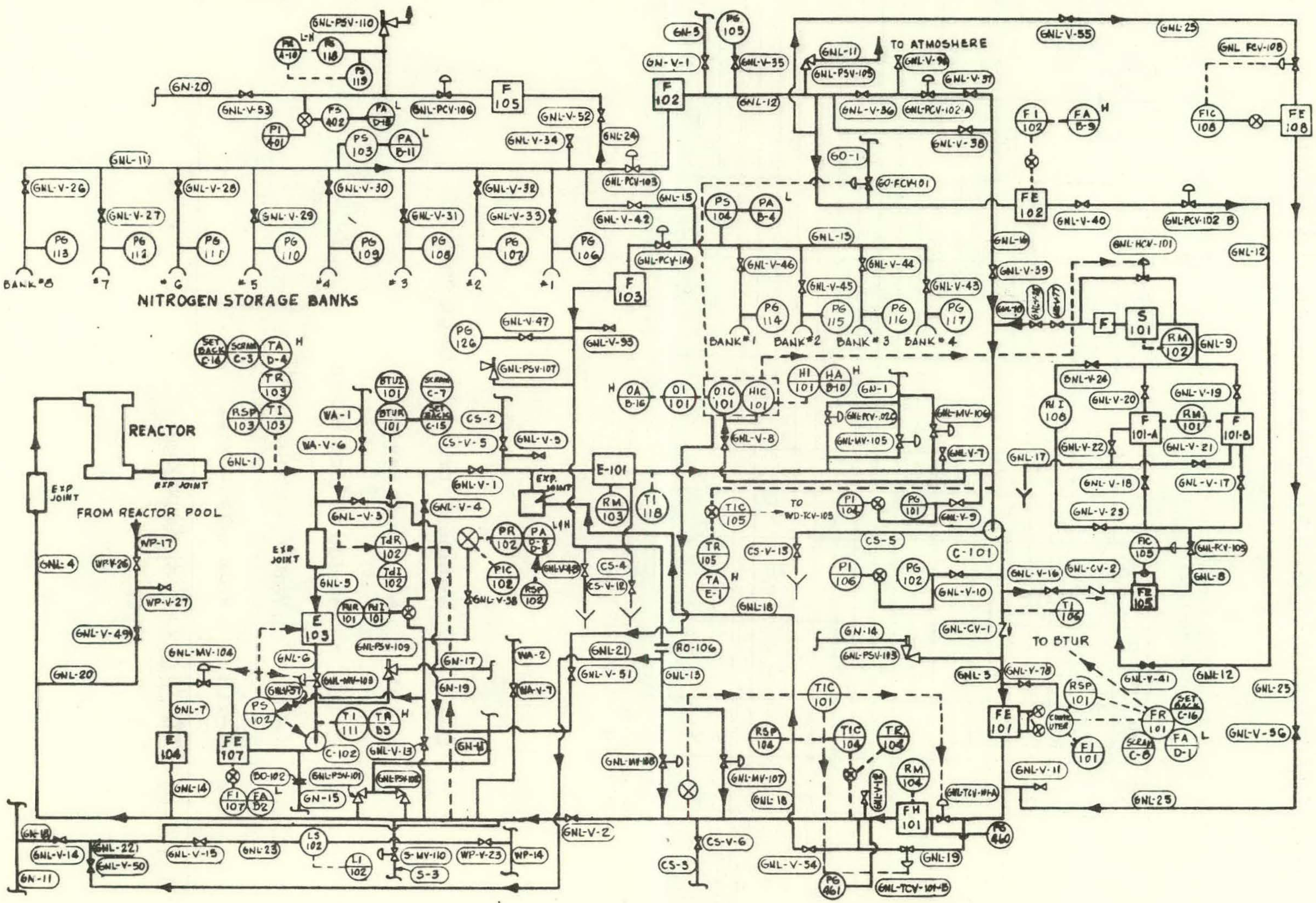
IDENTIFYING DESIGNATIONS FOR INSTRUMENTATION AND VALVES

PARAMETER

COMPONENT	FLOW F	LEVEL L	PRESSURE P	TEMPERA-TURE, T	ΔT Td	ΔP Pd	RADIATION R	HEAT OUT-PUT, Btu	OXYGEN O	HUMIDITY	CON-DUCTIVITY, C
ANNUNCIATOR - A	FA	LA	PA	TA					OA	HA	CA
CONTROLLER - C				TC							
CONTROL VALVE - CV*	FCV	LCV	PCV	TCV							
ELEMENT - E	FE	LE		TE							
GAGE - G			PG								
INDICATOR - I	FI	LI	PI	TI	TdI	PdI		BTUI		HI	
INDICATOR-CONTROLLER - IC	FIC	TIC	PIC	TIC					OIC	HIC	
METER - M											CM
MOTORIZED VALVE - MV*											
MONITOR - M							RM				
RECORDER - R	FR		PR	TR	TdR	PdR		BTUR			
RECORDER-CONTROLLER - RC			PRC								
RELIEF VALVE - RV*			PRV								
REMOTE SET POINT - RSP											
RESTRICTIVE ORIFICE - RO											
SAFETY VALVE - SV*			PSV								
SWITCH - S		LS	PS	TS							
TACHOMETER - TACH											
TEMPERATURE WELL - TW											
TRANSMITTER - T	FT	LT	PT								
MANUAL VALVE - V*			PV								

*VALVE DESIGNATIONS ARE USUALLY PRECEEDED BY INITIALS IDENTIFYING THE ASSOCIATED SYSTEM, AND FOLLOWED BY AN IDENTIFYING NUMBER, i.e., GNL-V-1, GN-MV-3, WD-LCV-202, etc.

E-103, 104	Gas-to-air heat exchangers, shutdown cooling loop
C-102	Shutdown cooling blower
E-101	Main loop (Gas-to-demineralized water)
C-101	Main loop compressor
FH-101	Oil-fired gas heater
S-101	Air filter-dryer assembly
F	Filter



MAIN NITROGEN LOOP

Report No. IDO-28598

WASTE AND MAKE-UP GAS SYSTEM - EQUIPMENT LEGEND

IDENTIFYING DESIGNATIONS FOR INSTRUMENTATION AND VALVES

PARAMETER

COMPONENT	FLOW F	LEVEL L	PRESSURE P	TEMPERA-TURE, T	ΔT Td	ΔP Pd	RADIATION R	HEAT OUT-PUT, Btu	OXYGEN O	HUMIDITY	CON-DUCTIVITY, C
ANNUNCIATOR - A	FA	LA	PA	TA					OA	HA	CA
CONTROLLER - C				TC							
CONTROL VALVE - CV*	FCV	LCV	PCV	TCV							
ELEMENT - E	FE	LE		TE							
GAGE - G			PG								
INDICATOR - I	FI	LI	PI	TI	TdI	PdI		BTUI		HI	
INDICATOR-CONTROLLER IC	FIC	TIC	PIC	TIC					OIC	HIC	
METER - M											CM
MOTORIZED VALVE - MV*											
MONITOR - M							RM				
RECORDER - R	FR		PR	TR	TdR	PdR		BTUR			
RECORDER-CONTROLLER - RC			PRC								
RELIEF VALVE - RV*			PRV								
REMOTE SET POINT - RSP											
RESTRICTIVE ORIFICE - RO											
SAFETY VALVE - SV*			PSV								
SWITCH - S		LS	PS	TS							
TACHOMETER - TACH											
TEMPERATURE WELL - TW											
TRANSMITTER - T	FT	LT	PT								
MANUAL VALVE - V*			PV								

* VALVE DESIGNATIONS ARE USUALLY PRECEEDED BY INITIALS IDENTIFYING THE ASSOCIATED SYSTEM, AND FOLLOWED BY AN IDENTIFYING NUMBER, i.e., GNL-V-1, GN-MV-3, WD-LCV-202, etc.

D-101	Demineralized water surge drum
C-101	Main loop compressor
C-102	Shutdown loop blower
T-201	Waste gas holder
C-201	Waste gas blower
BD	Burst diaphragm

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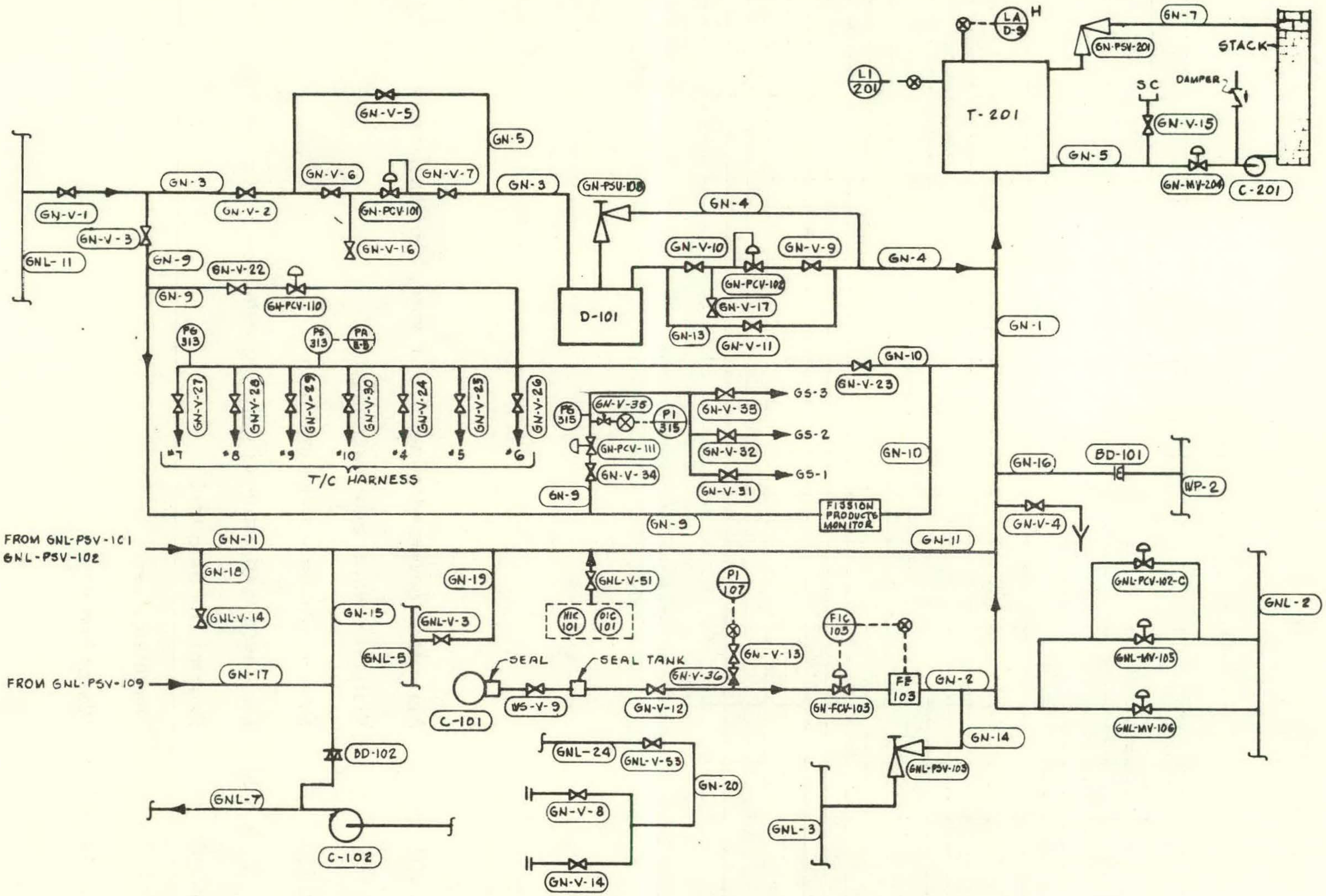

WASTE AND MAKE-UP GAS SYSTEM

Report No. IDO-28598

DEMINERALIZED WATER SYSTEM - EQUIPMENT LEGEND

IDENTIFYING DESIGNATIONS FOR INSTRUMENTATION AND VALVES

	PARAMETER										
COMPONENT	FLOW F	LEVEL L	PRESSURE P	TEMPERATURE, T	ΔT Td	ΔP Pd	RADIATION R	HEAT OUTPUT, Btu	OXYGEN O	HUMIDITY	CONDUCTIVITY, C
ANNUNCIATOR - A	FA	LA	PA	TA					OA	HA	CA
CONTROLLER - C				TC							
CONTROL VALVE - CV *	FCV	LCV	PCV	TCV							
ELEMENT - E	FE	LE		TE							
GAGE - G			PG								
INDICATOR - I	FI	LI	PI	TI	TdI	PdI		BTUI		HI	
INDICATOR-CONTROLLER - IC	FIC	TIC	PIC	TIC					OIC	HIC	
METER - M											CM
MOTORIZED VALVE - MV *											
MONITOR - M							RM				
RECORDER - R	FR		PR	TR	TdR	PdR		BTUR			
RECORDER-CONTROLLER - RC			PRC								
RELIEF VALVE - RV *			PRV								
REMOTE SET POINT - RSP											
RESTRICTIVE ORIFICE - RO											
SAFETY VALVE - SV *			PSV								
SWITCH - S		LS	PS	TS							
TACHOMETER - TACH											
TEMPERATURE WELL - TW											
TRANSMITTER - T	FT	LT	PT								
MANUAL VALVE - V *			PV								

* VALVE DESIGNATIONS ARE USUALLY PRECEEDED BY INITIALS IDENTIFYING THE ASSOCIATED SYSTEM, AND FOLLOWED BY AN IDENTIFYING NUMBER, i.e., GNL-V-1, GN-MV-3, WD-LCV-202, etc.

E-101	Main loop gas-to-demineralized water heat exchanger
E-102	Water-to-water heat exchanger
D-101	Demineralized water surge drum
P-101A & B	Demineralized water circulating pumps
D-201	Reactor flooding tank
S-301	Demineralizer
S-104	Water heater

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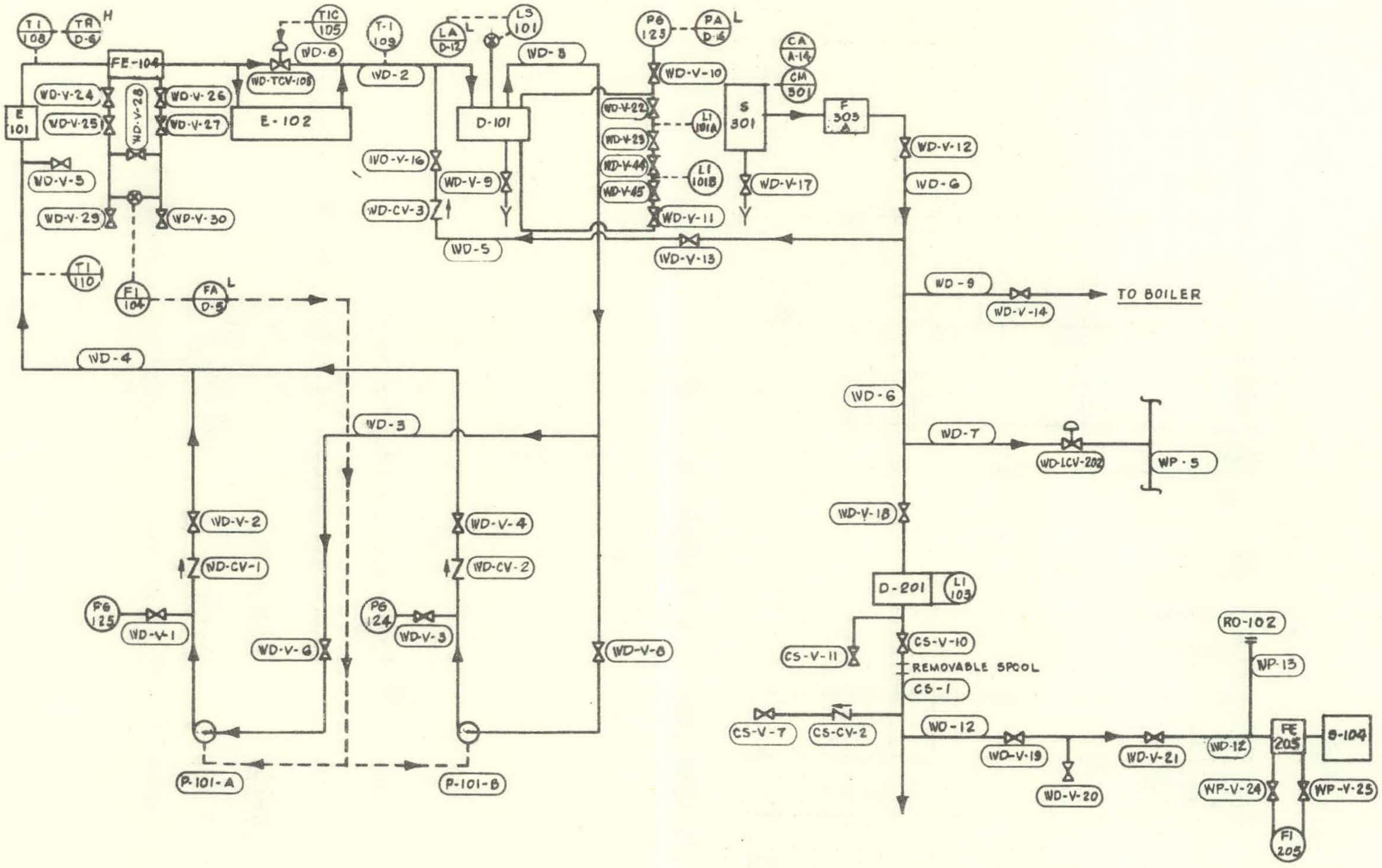


DEMINERALIZED WATER SYSTEM

D-7

Report No. IDO-28598

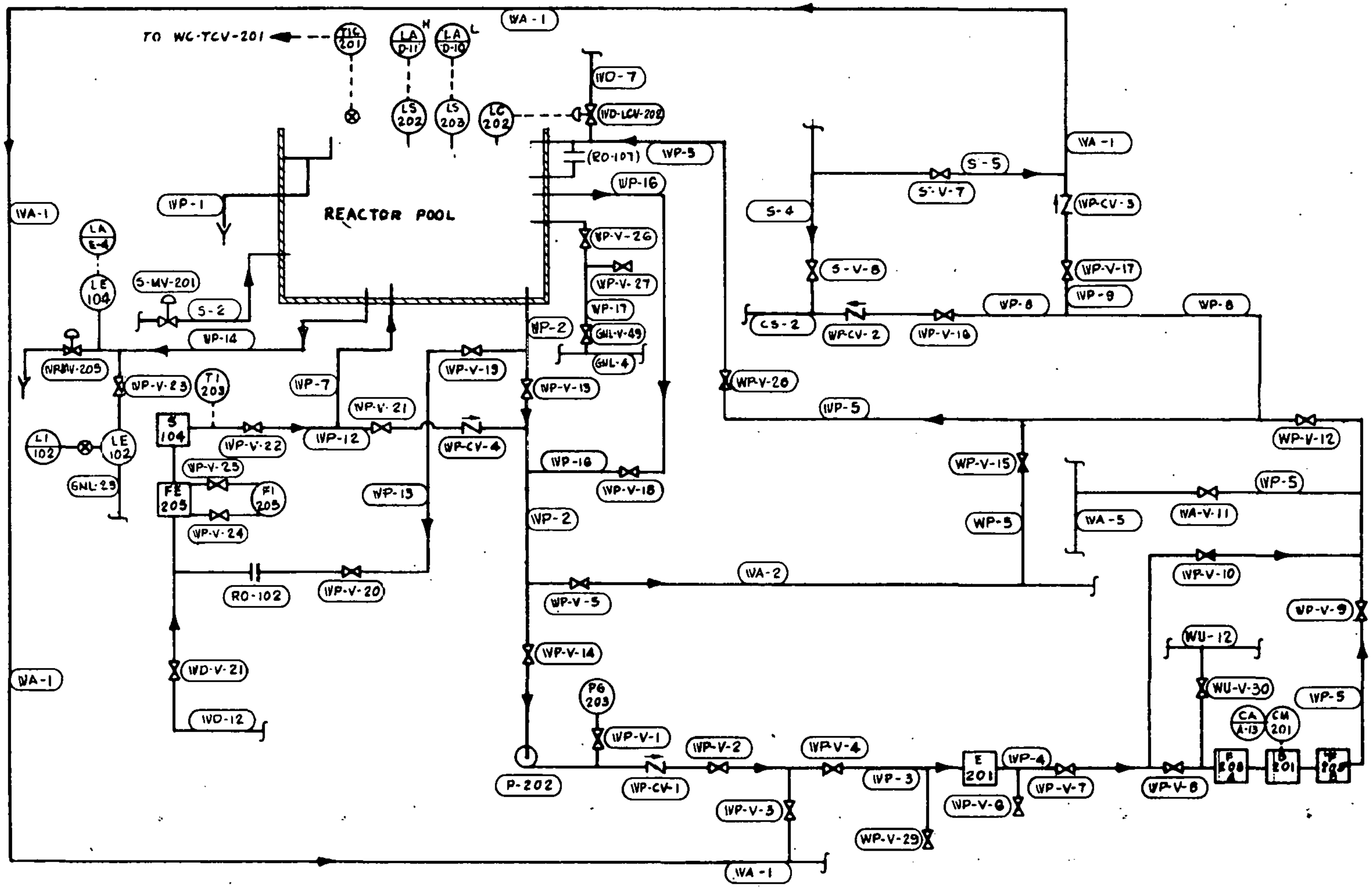
POOL WATER SYSTEM - EQUIPMENT LEGEND

IDENTIFYING DESIGNATIONS FOR INSTRUMENTATION AND VALVES

	PARAMETER										
COMPONENT	FLOW F	LEVEL L	PRESSURE P	TEMPERA- TURE, T	ΔT Td	ΔP Pd	RADIATION R	HEAT OUT- PUT, Btu	OXYGEN O	HUMIDITY	CON- DUCTIVITY, C
ANNUNCIATOR - A	FA	LA	PA	TA					OA	HA	CA
CONTROLLER - C				TC							
CONTROL VALVE - CV*	FCV	LCV	PCV	TCV							
ELEMENT - E	FE	LE		TE							
GAGE - G			PG								
INDICATOR - I	FI	LI	PI	TI	TdI	PdI		BTUI		HI	
INDICATOR-CONTROLLER - IC	FIC	TIC	PIC	TIC					OIC	HIC	
METER - M											CM
MOTORIZED VALVE - MV*											
MONITOR - M							RM				
RECORDER - R	FR		PR	TR	TdR	PdR		BTUR			
RECORDER-CONTROLLER - RC			PRC								
RELIEF VALVE - RV*			PRV								
REMOTE SET POINT - RSP											
RESTRICTIVE ORIFICE - RO											
SAFETY VALVE - SV*			PSV								
SWITCH - S		LS	PS	TS							
TACHOMETER - TACH											
TEMPERATURE WELL - TW											
TRANSMITTER - T	FT	LT	PT								
MANUAL VALVE - V*			PV								

* VALVE DESIGNATIONS ARE USUALLY PRECEEDED BY INITIALS IDENTIFYING THE ASSOCIATED SYSTEM, AND FOLLOWED BY AN IDENTIFYING NUMBER, i.e., GNL-V-1, GN-MV-3, WD-LCV-202, etc.

P-202	Pool water circulating pump
E-201	Pool water-to-cooling water heat exchanger
S-201	Demineralizer
S-104	Water heater



POOL WATER SYSTEM

LIQUID WASTE SYSTEM - EQUIPMENT LEGEND

IDENTIFYING DESIGNATIONS FOR INSTRUMENTATION AND VALVES

PARAMETER

COMPONENT	FLOW F	LEVEL L	PRESSURE P	TEMPERA-TURE, T	ΔT Td	ΔP Pd	RADIATION R	HEAT OUT-PUT, Btu	OXYGEN O	HUMIDITY	CON-DUCTIVITY, C
ANNUNCIATOR - A	FA	LA	PA	TA					OA	HA	CA
CONTROLLER - C				IC							
CONTROL VALVE - CV*	FCV	LCV	PCV	TCV							
ELEMENT - E	FE	LE		TE							
GAGE - G			PG								
INDICATOR - I	FI	LI	PI	TI	TdI	PdI		BTUI		HI	
INDICATOR-CONTROLLER - IC	FIC	TIC	PIC	TIC					OIC	HIC	
METER - M											CM
MOTORIZED VALVE - MV*											
MONITOR M							RM				
RECORDER - R	FR		PR	TR	TdR	PdR		BTUR			
RECORDER-CONTROLLER - RC			PRC								
RELIEF VALVE - RV*			PRV								
REMOTE SET POINT - RSP											
RESTRICTIVE ORIFICE - RO											
SAFETY VALVE - SV*			PSV			–					
SWITCH - S		LS	PS	TS							
TACHOMETER - TACH											
TEMPERATURE WELL - TW											
TRANSMITTER - T	FT	LT	PT								
MANUAL VALVE - V*			PV								

*VALVE DESIGNATIONS ARE USUALLY PRECEEDED BY INITIALS IDENTIFYING THE ASSOCIATED SYSTEM, AND FOLLOWED BY AN IDENTIFYING NUMBER, i.e., GNL-V-1, GN-MV-3, WD-LCV-202, etc.

D-202, 203	Liquid waste storage tanks
P-203	Sump pump
P-205	Liquid waste transfer pump
P-206	Reactor loop wash pump
E-102	Shutdown cooling loop blower
S-203	Mixing chamber
S-203	Backflow preventer

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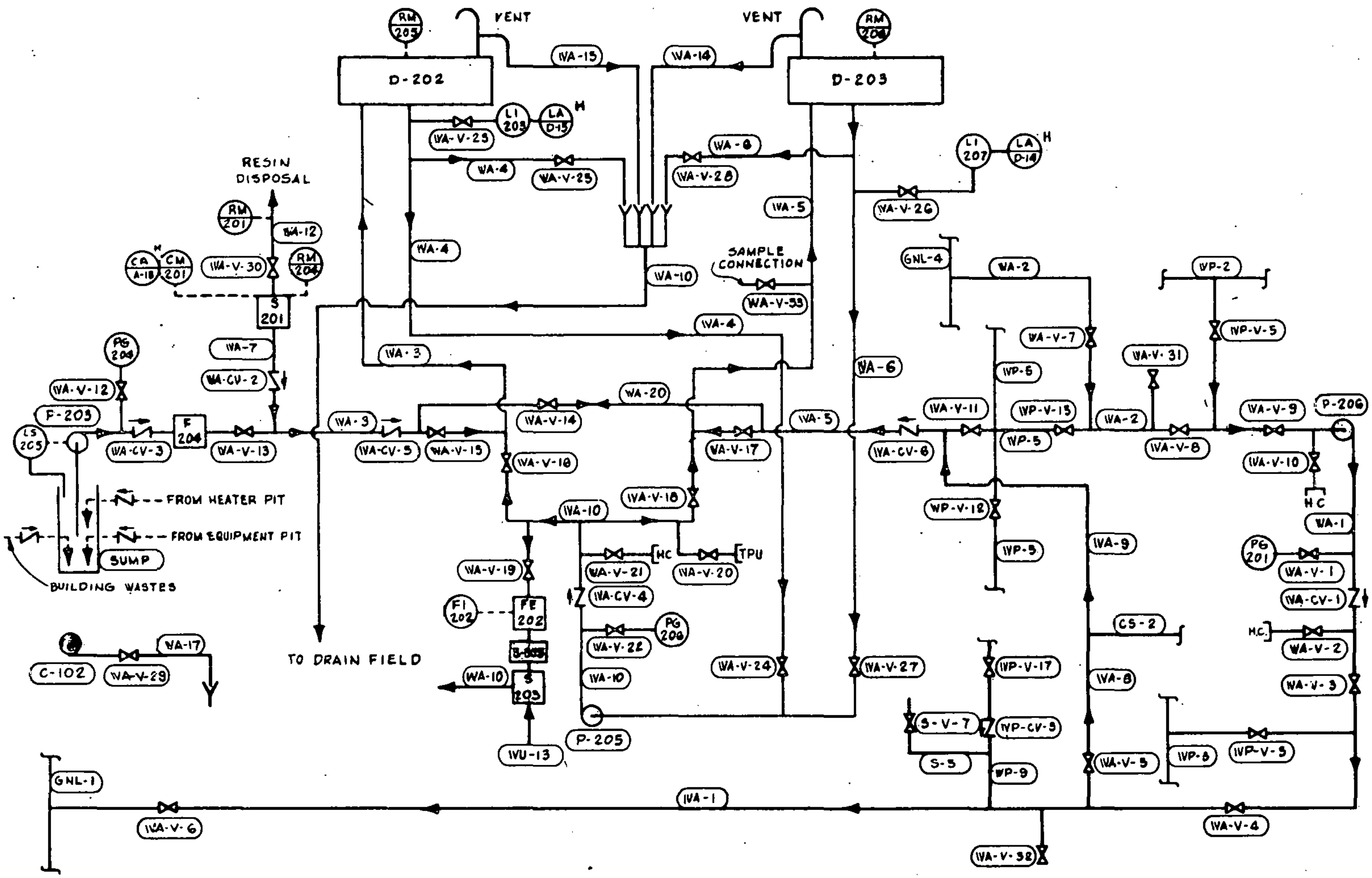

LIQUID WASTE SYSTEM

COOLING WATER SYSTEM - EQUIPMENT LEGEND

IDENTIFYING DESIGNATIONS FOR INSTRUMENTATION AND VALVES

PARAMETER

COMPONENT	FLOW F	LEVEL L	PRESSURE P	TEMPERA-TURE, T	ΔT Td	ΔP Pd	RADIATION R	HEAT OUT-PUT, Btu	OXYGEN O	HUMIDITY	CON-DUCTIVITY, C
ANNUNCIATOR - A	FA	LA	PA	TA					OA	HA	CA
CONTROLLER - C				TC							
CONTROL VALVE - CV *	FCV	LCV	PCV	TCV							
ELEMENT - E	FE	LE		TE							
GAGE - G			PG								
INDICATOR - I	FI	LI	PI	TI	TdI	PdI		BTUI		HI	
INDICATOR-CONTROLLER - IC	FIC	TIC	PIC	TIC					OIC	HIC	
METER - M											CM
MOTORIZED VALVE - MV *											
MONITOR - M							RM				
RECORDER - R	FR		PR	TR	TdR	PdR		BTUR			
RECORDER-CONTROLLER - RC			PRC								
RELIEF VALVE - RV *			PRV								
REMOTE SET POINT - RSP											
RESTRICTIVE ORIFICE - RO											
SAFETY VALVE - SV *			PSV								
SWITCH - S		LS	PS	TS							
TACHOMETER TACH											
TEMPERATURE WELL - TW											
TRANSMITTER - T	FT	LT	PT								
MANUAL VALVE - V *			PV								

* VALVE DESIGNATIONS ARE USUALLY PRECEEDED BY INITIALS IDENTIFYING THE ASSOCIATED SYSTEM, AND FOLLOWED BY AN IDENTIFYING NUMBER, i.e., GNL-V-1, GN-MV-3, WD-LCV-202, etc.

E-201	Pool Water-to-cooling water heat exchanger
E-102	Demineralized water-to-cooling water heat exchanger
C-101	Main loop compressor
S-101	Main loop gas dryer
S-303	Instrument air dryer
C-301	Utility air compressor
E-301	Utility air aftercooler
C-302	Instrument air compressor
E-302	Instrument air aftercooler
S-304	Diesel generator
CT-301	Cooling tower
S-302	Chlorinator
P-304A & B	Cooling water circulating pumps

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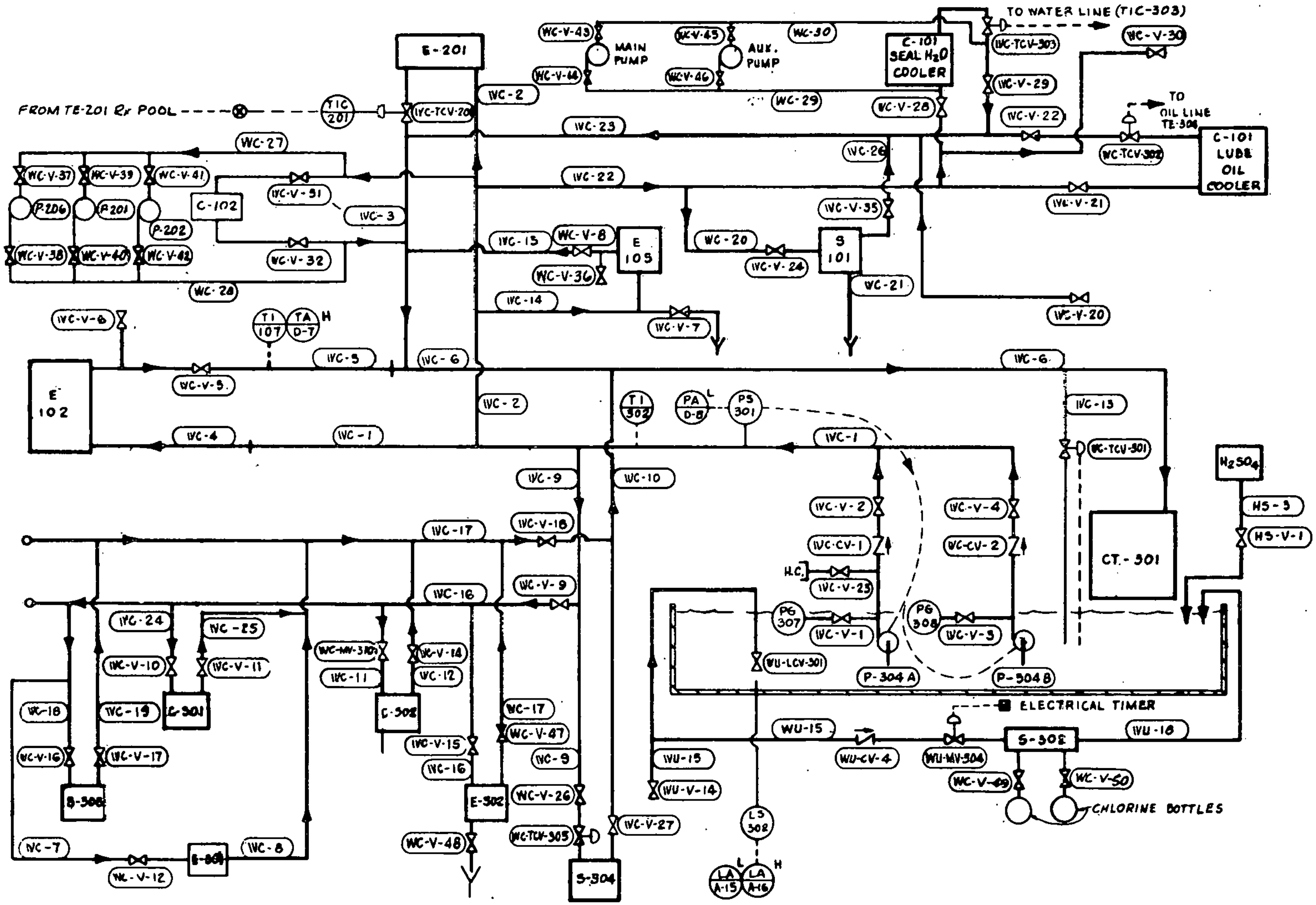


COOLING WATER SYSTEM

UTILITY WATER SYSTEM - EQUIPMENT LEGEND

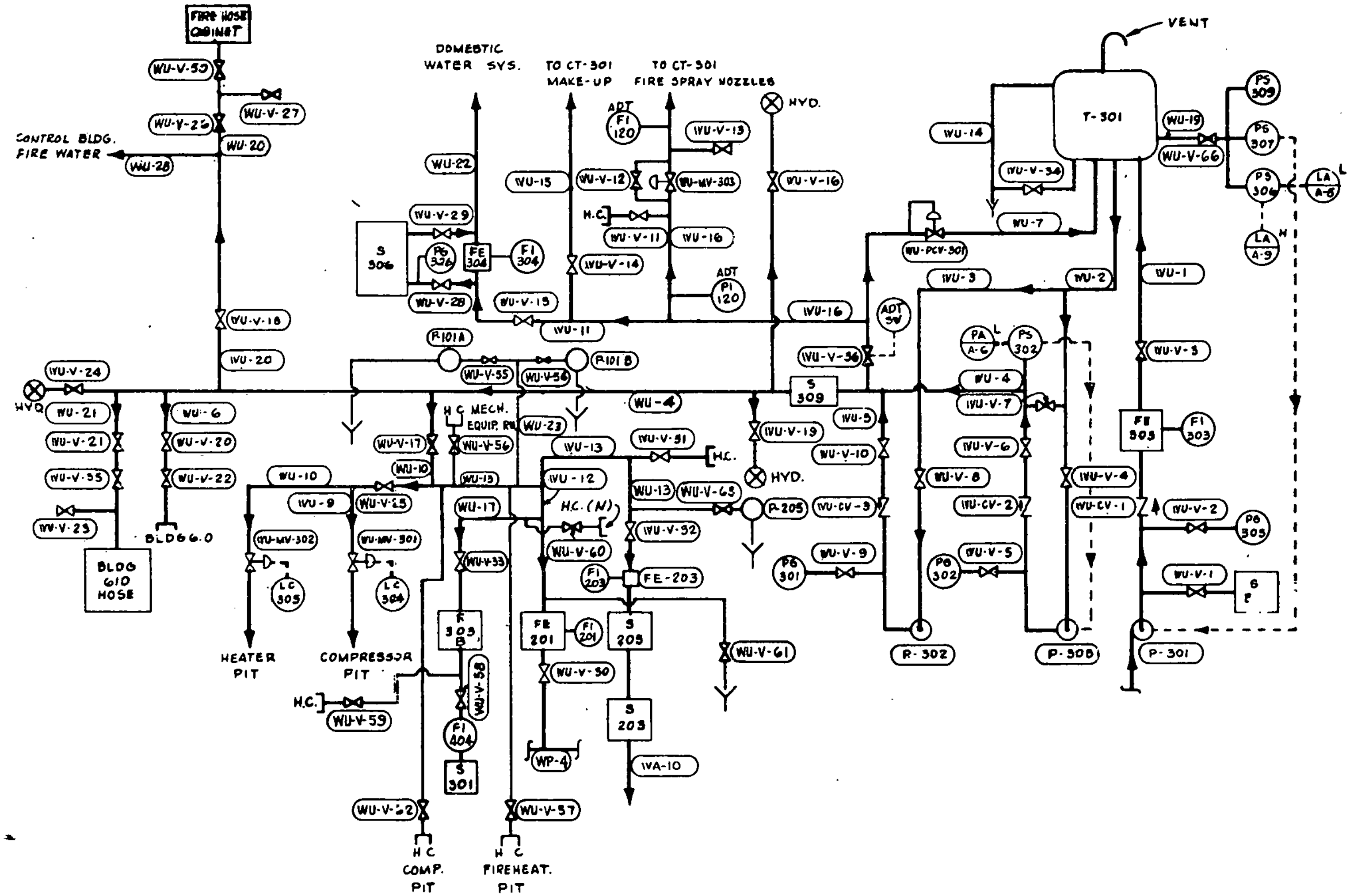
IDENTIFYING DESIGNATIONS FOR INSTRUMENTATION AND VALVES

PARAMETER

COMPONENT	FLOW F	LEVEL L	PRESSURE P	TEMPERATURE, T	ΔT Td	ΔP Pd	RADIATION R	HEAT OUTPUT, Btu	OXYGEN O	HUMIDITY	CONDUCTIVITY, C
ANNUNCIATOR - A	FA	LA	PA	TA					OA	HA	CA
CONTROLLER - C				TC							
CONTROL VALVE - CV*	FCV	LCV	PCV	TCV							
ELEMENT - E	FE	LE		TE							
GAGE - G			PG								
INDICATOR - I	FI	LI	PI	TI	TdI	PdI		BTUI		HI	
INDICATOR-CONTROLLER - IC	FIC	TIC	PIC	TIC					OIC	HIC	
METER - M											CM
MOTORIZED VALVE - MV*											
MONITOR - M							RM				
RECORDER - R	FR		PR	TR	TdR	PdR		BTUR			
RECORDER-CONTROLLER - RC			PRC								
RELIEF VALVE - RV*			PRV								
REMOTE SET POINT - RSP											
RESTRICTIVE ORIFICE - RO											
SAFETY VALVE - SV*			PSV								
SWITCH - S		LS	PS	TS							
TACHOMETER - TACH											
TEMPERATURE WELL - TW											
TRANSMITTER - T	FT	LT	PT								
MANUAL VALVE - V*			PV								

* VALVE DESIGNATIONS ARE USUALLY PRECEEDED BY INITIALS IDENTIFYING THE ASSOCIATED SYSTEM, AND FOLLOWED BY AN IDENTIFYING NUMBER, i.e., GNL-V-1, GN-MV-3, WD-LCV-202, etc.

S-301	Demineralizer
S-203	Mixing chamber
P-101A & B	Demineralized water circulating pumps
S-309 & S-205	Backflow preventers
P-205	Liquid waste transfer pump
P-302	Utility water pump
P-303	Fire protection pump
P-301	Well pump
S-308	Vacuum breaker
T-301	Utility water storage tank
S-306	Domestic water chlorinator
CT-301	Cooling tower
Hyd	Fire hydrant
He	Hose connection



UTILITY WATER SYSTEM

UTILITY AND INSTRUMENT AIR SYSTEMS - EQUIPMENT LEGEND

IDENTIFYING DESIGNATIONS FOR INSTRUMENTATION AND VALVES

COMPONENT	PARAMETER FLOW F	LEVEL L	PRESSURE P	TEMPERA-TURE, T	ΔT Td	ΔP Pd	RADIATION R	HEAT OUT-PUT, Btu	OXYGEN O	HUMIDITY	CON-DUCTIVITY, C
ANNUNCIATOR - A	FA	LA	PA	TA					OA	HA	CA
CONTROLLER - C				TC							
CONTROL VALVE - CV*	FCV	LCV	PCV	TCV							
ELEMENT - E	FE	LE		TE							
GAGE - G			PG								
INDICATOR - I	FI	LI	PI	TI	TdI	PdI		BTUI		HI	
INDICATOR-CONTROLLER - IC	FIC	TIC	PIC	TIC					OIC	HIC	
METER - M											CM
MOTORIZED VALVE - MV*											
MONITOR - M							RM				
RECORDER - R	FR		PR	TR	TdR	PdR		BTUR			
RECORDER-CONTROLLER - RC			PRC								
RELIEF VALVE - RV*			PRV								
REMOTE SET POINT - RSP											
RESTRICTIVE ORIFICE - RO											
SAFETY VALVE - SV*			PSV								
SWITCH - S		LS	PS	TS							
TACHOMETER - TACH											
TEMPERATURE WELL - TW											
TRANSMITTER - T	FT	LT	PT								
MANUAL VALVE - V*			PV								

*VALVE DESIGNATIONS ARE USUALLY PRECEEDED BY INITIALS IDENTIFYING THE ASSOCIATED SYSTEM, AND FOLLOWED BY AN IDENTIFYING NUMBER, i.e., GNL-V-1, GN-MV-3, WD-LCV-202, etc.

B	Burner
BCU	Burner control unit
C-302	Instrument air cooler
D-302	Instrument air receiver
S-303	Instrument air dryer
C-301	Utility air compressor
E-302	Utility air receiver
FH-101	Oil-fired gas heater
S-301	Demineralizer

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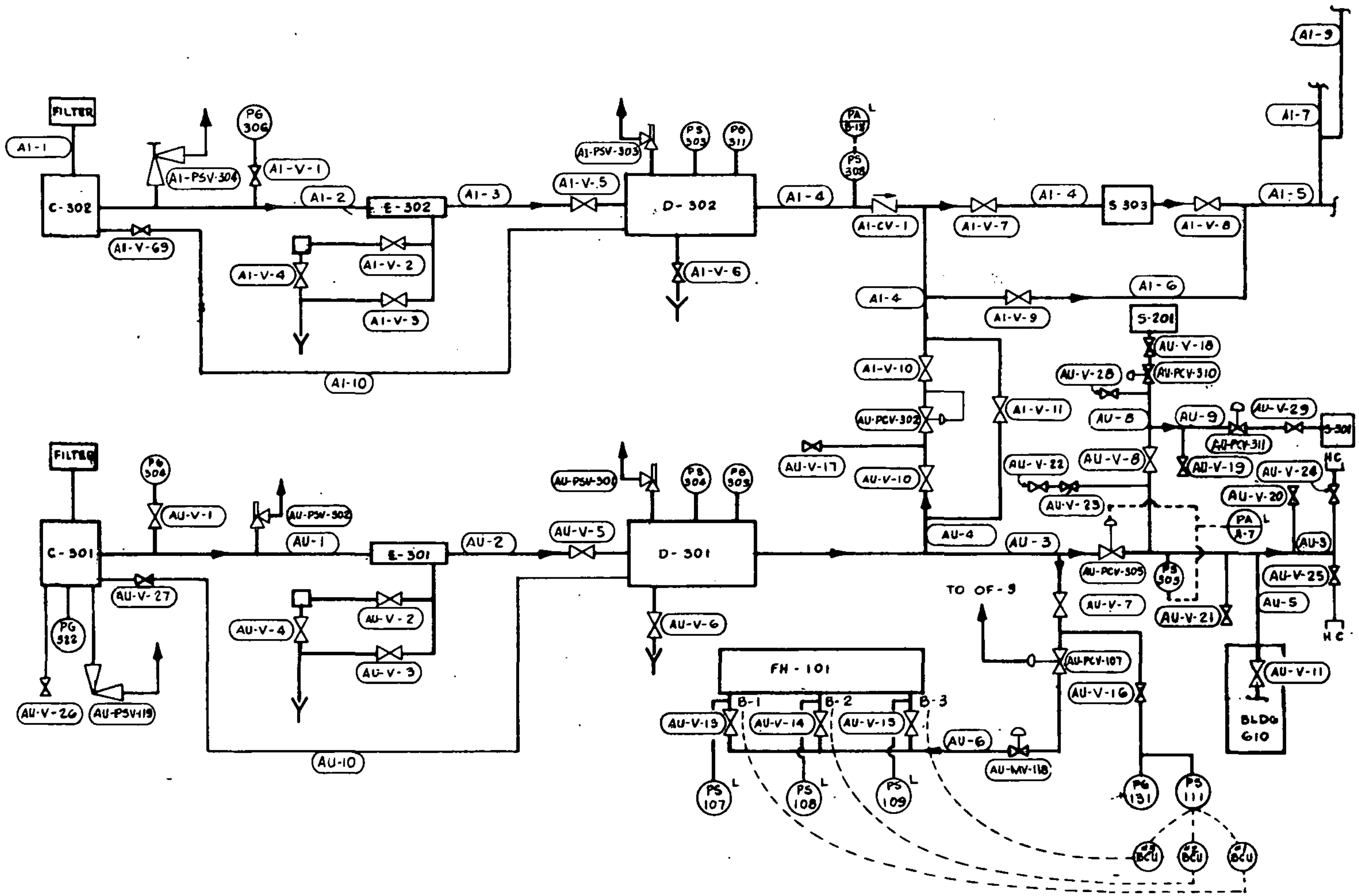

UTILITY AND INSTRUMENT AIR SYSTEMS

APPENDIX E

TYPICAL GCRE-I EXPERIMENTAL PROCEDURE

This appendix presents a typical ANSOP as used in the GCRE-I experimental program.

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STANDARD OPERATING PROCEDURE
GAS-COOLED REACTOR EXPERIMENT-I

PROCEDURE SECTION 9 EXPERIMENTAL PROCEDURE

TITLE: INITIAL CRITICAL EXPERIMENT WITH THE STAINLESS STEEL TUBE BUNDLE

SCOPE

This procedure outlines the steps to be followed in the initial determination of the cold critical loading of the reactor with a stainless steel tube bundle. The procedure terminates with that core loading which is less than one full fuel element (peripheral position) supercritical.

The initial critical loading will be determined in the wet condition. The critical assembly calculations have indicated that a symmetrical loading of approximately 39 IZ fuel elements will be required to attain wet criticality. After proper preparation, this procedure specifies incremental additions of reactivity (by adding fuel elements in specified locations) until criticality is attained. Inverse multiplication plots and inverse count rates are maintained throughout this procedure to permit evaluation of the progress and to predict the final critical loading. These plots are compared with previous data and predicted plots of inverse multiplication to anticipate non-standard conditions.

PROCEDURE

1. Prerequisites for the initiation of this procedure are:

a. Only AGC authorized personnel in the area.

b. Check out of the facility and equipment as required for the completion of this procedure.

c. Exclusion area established and guard force on duty.

d. Master Emergency Plan (ANSOP 3900) in effect.

NOTE: Access to the control room during this procedure is limited to those personnel whose duties require such access.

2. Personnel required for this test shall include at least the following:

Operations Manager

Operating Superintendent

Health and Safety Supervisor

Nuclear Engineer

Shift Supervisor

Three Technicians

3. A temporary start-up control center shall be roped off to outline the physical limits in the control room as follows:

a. Provide necessary desks, tables, etc., to accommodate the equipment and provide work space for plotting and calculations.

b. Provide an electric interval timer capable of readings to the nearest second.

c. Provide a calculating machine.

4. Install temporary start-up counting circuitry as follows:

a. Insert water-tight aluminum thimbles into positions B-1, H-1, and H-8.

b. Install BF_3 detectors in the aluminum thimbles so that the chamber midplane is coincident with the midplane of the reactor.

c. Connect the detectors to scalers located in the control room.

5. Set the scram microswitch on the Log-N recorder at 0.005 on the scale.

6. Prepare the Fuel Element Handling Schedule (ANSOL 5758).

7. Prepare an SWP for charging the source and fuel elements into the reactor.

8. Check that the pool is filled to within one inch of the lower side of the upper plenum, upper flange.

9. Check that the reactor is filled to the top of the fuel element holddowns with demineralized water. Fill into GNL-1 through WA-V-6 from the pool water system.

10. Insert the startup source in its holder into position B-8.

11. Shield or position BF_3's to attain a count rate of approximately 10 times the background count.

    NOTE: The instrument supervisor will check the operation of the scalers and the BF_3's for reliability.

12. Check the neutron detection system per ANSOP 5709.

13. Check the "Water in Rods" indicating system per ANSOP 5727.

14. Complete the safety circuit check per ANSOP 5707.

15. Check the operation of the control rod pressure system per ANSOP 5735.

16. Complete the rod operability check per ANSOP 5710.

17. Assign personnel as follows:

Function	Classification
Fuel Element Handlers	Technicians
Console Operator	Technician
Control Center Supervisor	Shift Supervisor
Scaler Operator	Technician
Plotter	Technician

NOTES:

1) From this point until the end of the procedure the Control Center supervisor will complete ANSOL 5727.

2) During this period, the Operations Manager, the Operating Superintendent, the Health and Safety Supervisor, and the Nuclear Engineer will observe the work and monitor the procedure. Any one of these is authorized to stop work at any time. Operations Manager must authorize continuance after such stoppage. One of the above four persons must be in the Control Center at all times.

3) If the procedure from this point to criticality is interrupted for longer than four hours, repeat steps 12 through 16 before proceeding.

18. Make up the interlock circuit and energize rod magnets.

19. Withdraw Gang I to full-out position.

20. Move five fuel elements (according to ANSOL 5758) from the Waste Water Pump house to the reactor pool in the following steps:

a. Check that the elements have been wiped clean.

b. Load the elements on the transfer rack and fasten securely.

c. Move the transfer rack on the 3/4 ton truck to the reactor building through doors 32 and 30 respectively.

d. Move the transfer rack from the 3/4 ton truck to the pool edge.

21. Charge the five elements into positions D-8, E-8, F-8, G-7, and H-6 in accordance with this procedure:

a. Move the transfer rack to the crane basket.

b. Lower the basket with two fuel element handlers into the reactor pit and rest the basket on the four instrument flanges.

c. Pick up a fuel element by hand and charge it part way into the proper reactor position. (Fuel element handler #1).

d. Engage the handling tool, lower the element into position, release and retract the handling tool. (Fuel element handler #2).

e. Verify that the fuel element has been charged into the proper position and initial ANSOL 5758. (Fuel element handler #1).

f. Verify that the element has been charged into the proper position (Control Center Supervisor).

22. Clear reactor area of all personnel.

23. Take 3 one minute counts with a one minute wait between counts on the BF_3 scalers. Record the counting data on ANSOL 5726.

24. Complete ANSOL 5726 and plot the data.

25. Repeat steps 20 through 24 by charging the elements five at a time into the following ordered positions until 30 elements have been charged.

Element	Position	Element	Position
6	G-6	19	C-5
7	F-7	20	B-5
8	E-7	21	B-4
9	D-7	22	D-5
10	C-7	23	C-4
11	B-6	24	B-3
12	C-6	25	C-3
13	D-6	26	D-4
14	E-6	27	D-3
15	F-6	28	E-4
16	H-5	29	F-5
17	G-5	30	E-3
18	E-5		

26. Withdraw Gang II to the full out position.

27. After one minute wait,withdraw the setback rod.

28. After a one minute wait, withdraw the shim gang until the "Rod In" lights go out.

29. Withdraw the fine rod to the full out position.

30. After a one minute wait, withdraw the shim gang to the full out position.

31. Repeat steps 23 and 24.

    NOTE: The neutron count at this point will be the base count for succeeding 1/M calculations. Check that the scalers have a capacity of at least three more decades before saturation.

32. Insert the fine and setback rods.

33. Repeat step 20 with one fuel element.

34. Charge the fuel element into position F-4 per step 21 (element 31).

35. Repeat step 22.

36. Withdraw the fine and setback rods with a one minute wait between each rod withdrawal.

37. Repeat steps 23 and 24.

38. Repeat step 20 with one fuel element.

39. Charge the fuel element into position G-4 per step 21 (element 32).

40. Repeat steps 22, 23, and 24.

41. Withdraw the setback rod.

42. Repeat steps 23 and 24.

43. Withdraw the fine rod.

44. Repeat steps 23 and 24.

45. Insert the fine and setback rods.

46. Repeat step 20 with one fuel element.

47. Charge the fuel element into position H-4 per step 21 (element 33).

48. Repeat steps 22 and 36.

49. Repeat steps 23 and 24.

50. Insert the fine and setback rods.

51. Repeat step 20 with one fuel element.

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52. Charge the fuel element into position E-2 per step 21 (element 34).

53. Repeat steps 22, 23 and 24.

54. Withdraw the setback rods.

55. Repeat steps 23 and 24.

56. Withdraw the fine rod.

57. Repeat steps 23 and 24.

58. Insert the fine and setback rods.

59. Repeat step 20 with one fuel element.

60. Charge the fuel element into position F-3 per step 21 (element 35).

61. Repeat steps 22 and 36.

62. Repeat steps 23 and 24.

63. Insert the fine and setback rods.

64. Repeat step 20 with one fuel element.

65. Charge the fuel element into position G-3 per step 21 (element 36).

66. Repeat steps 22, 23 and 24.

67. Withdraw the setback rod.

68. Repeat steps 23 and 24.

69. Withdraw the fine rod.

70. Repeat steps 23 and 24.

71. Insert the fine and setback rods.

72. Repeat steps 64 through 71 by charging one element at a time into the following order positions until criticality is attained.

Element	Position	Element	Position
37	H-3	44	G-2
38	C-8	45	H-2
39	B-7		
40	B-2		
41	C-2		
42	G-8		
43	H-7		

CAUTION: Beginning with the loading of the element into position H-3 or the attainment of a 1/M of 0.2 whichever occurs first, the following additional steps will be taken:

a. The operating supervisor will obtain permission from the Operations Manager before initiating the sequence of steps to accomplish the loading of each next element in series.

b. All personnel will be continually alert for the attainment of criticality as evidenced by one or more of the following:

1) 1/M of essentially zero.

2) Sustained period on the period recorder or a sustained rise on the count rate or Log-N recorders.

Any of the foregoing indications or any other unusual occurrence shall be reported to the operating supervisor.

c. Adjust the shim gang, if necessary, to attain criticality on the fine rod.

73. Once criticality is attained, insert the fine rod.

CAUTION: Do not allow the power to rise above 0.005 on the Log-N recorder.

74. Run all other rods in unless otherwise directed by the Operations Manager.

APPENDIX F

TYPICAL GCRE-I TECHNICAL STANDARDS

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TECHNICAL STANDARDS
GAS-COOLED REACTOR EXPERIMENT-I

SYSTEM OR EQUIPMENT

Primary Coolant Loop and Associated Equipment

STANDARDS

1. Flow

   a. The system shall be operated in such fashion that the flow as measured by the mass flow meter does not exceed 20 lbs/sec.

   b. A circulation rate of at least 1.7 lbs/sec shall be maintained at all times when the reactor power level, as indicated by any power level indicator, is greater than 5,000 watts.

2. Temperature

   a. The temperature of the circulating coolant stream shall never be permitted to rise above 1250^{o}F at any location downstream of the fired heater and upstream of the main heat exchanger except that higher temperatures are permitted at fuel element effluent sensing locations.

   c. The temperature of the circulating coolant stream shall never be permitted to rise above 200^{o}F unless the pool water level is above the top of the reactor pressure vessel.

3. Composition

a. The gas circulating in the primary coolant loop shall have the following composition before the reactor power is raised above 20 kw as indicated on any power level indicator.

	Min.	Max.
Nitrogen and Oxygen	99.9%	---
Oxygen	0.1%	1.0%
Humidity (dewpoint)	---	$-6^{\circ}F$

4. Pressure

a. The operating pressure at any point in the reactor shall not exceed 220 psig.

b. The reactor may be pressurized to 250 psig for pressure testing provided no fuel is present in the reactor and provided the temperature of the inlet coolant gas does not exceed $800^{\circ}F$.

c. The operating pressure at any point in the circulating coolant loop exclusive of the reactor shall not exceed 330 psig.

5. Activity

a. The primary coolant system shall not be operated when the radioactivity level as measured by any of the RAM detectors exceeds 50 R/hr.

6. E-101 Main Heat Exchanger

a. This equipment shall be operated to conform to the following standards:

1) The exit gas temperature shall not exceed $250^{\circ}F$.

2) The exit water temperature shall not exceed $250^{\circ}F$.

3) Inlet gas temperature shall not exceed $250^{\circ}F$ without normal water circulation.

7. C-101 Primary Coolant Compressor

a. The compressor shall be operated to conform to the following standards:

1) The gas temperature at the suction shall not exceed 250^{o}F.

2) The compressor shall not be rotated by motor unless the seal water supply pressure (PI-304) is at least 20 psig.

3) The compressor shall not be rotated by motor unless the lubricating oil supply pressure (PG-314) is at least 5 psig.

4) The seal water supply pressure (PI-304) shall be 8-12 psi higher than the gas suction pressure, except as specified in 2) above.

5) The back pressure on the seal system (PI-107) shall be maintained by adjusting FIC-103 to pass not less than 4 lbs/hr. The back pressure shall always be less than the suction pressure but has no fixed value.

6) The compressor bearing temperature (TI-121) shall not exceed 180^{o}F. The compressor bearing temperature shall not exceed 170^{o}F unless the compressor operation is closely supervised.

b. The seal water system shall be operated to conform to the following standards:

1) The seal water reservoir shall be filled with demineralized water and maintained with a pH value of 6.5 - 8.0.

2) The seal water reservoir shall contain at least 52 gallons of water, a level about 8¼ inches up from the bottom.

3) The seal water supply pressure (PG-312) downstream of filter shall be at least 220 psig.

4) The seal water supply temperature (TI-305) shall not exceed 120^{o}F.

5) The pressure drop across the seal water filter shall not exceed 10 psig.

6) The water pressure at any location shall not exceed 350 psig.

7) The oil concentration in the seal water shall not exceed 20 ppm.

c. The lubricating oil system shall be operated to conform to the following standards:

1) The lube oil reservoir shall be filled with clean filtered oil having a viscosity of 300 SSU at 100°F and a viscosity index of 100.

2) The lube oil reservoir shall not contain less than 58 gallons of oil, a level about 21¼ inches down along the oil stick or about 7 3/4 inches up from the bottom.

3) The lube oil supply pressure (PI-302) downstream of filter shall be at least 10 psig.

4) The lube oil supply temperature (TI-120) shall not exceed 125°F.

5) The lube oil return temperatures (TI-116 and TI-118) shall not exceed 170°F and shall not exceed 160°F unless the compressor operation is closely supervised.

6) The pressure drop across the lube oil filter shall not exceed 10 psig.

7) The oil pressure at any location shall not exceed 80 psig.

d. The gear unit shall be operated to conform to the following standards:

1) The gear unit shall not be rotated by motor unless the lubricating oil supply pressure at PI-303 is at least 5 psig.

3. The emergency motor shall be operated to conform to the following standards:

1) The emergency motor shall not be operated simultaneously with the main motor.

2) The emergency motor oil wells shall be filled with clean filtered oil having a viscosity of 300 SSU at 100°F.

f. The Dynamatic coupling shall be operated to conform to the following standards:

1) The cooling water for the coupling shall have a pH value range of 7-9 and should not have a mineral content exceeding the amounts tabulated below:

Total carbonate	500 ppm
Total sulphate	300 ppm
Total chloride	50 ppm
Total iron	75 ppm
Total alumina	75 ppm

2) The cooling water supply pressure shall be at least 35 psig.

3) The cooling water supply temperature shall be less than 90°F.

4) The water temperature at the discharge outlet of the coupling should not exceed 160°F.

5) The discharge water must flow freely, by gravity, away from the unit, from the water outlet in the coupling. The water piping must not impose restrictions on the flow of this discharge water or flooding of the unit may result.

6) The several vents in the housing shall not be obstructed.

7) The input and output end bearings and the pilot bearings shall be lubricated using a lithium soap based grease such as Standard Oil (California) R.P.M. Aviation No. 5 or an equivalent.

g. The main motor shall be operated to conform to the following standards:

1) The main motor shall not be operated simultaneously with the emergency motor.

2) The main motor oil wells shall be filled with clean filtered oil having a viscosity of 300 SSU at 100°F.

8. FH-101 Fired Heater

a. This equipment shall be operated to conform to the following standards:

1) Tube wall temperatures shall not exceed 1275°F.

2) The atomizing air pressure shall be at least 100 psig.

3) The fuel oil pressure shall be at least 115 psig.

9. F-101 A & B Gas Filters

a. This equipment shall be operated to conform to the following standards:

1) The flow rate to the filters (controlled by FIC-105) shall not exceed 135 SCFM.

2) The gas temperature shall not exceed 200°F.

3) The filter shall be changed when the flow, as indicated by FIC-105, is reduced to 55 SCFM or when the activity, as shown by RM-101, reaches 10 R/hr, whichever occurs first.

10. C-102 Shutdown Cooling Blower

a. This equipment shall be operated to conform to the following standards:

1) The pressure, as indicated by PR-102, in the reactor inlet duct shall not exceed 15 psig.

2) The inlet temperature, as indicated by TI-111, shall not exceed 450°F.

11. E-103 Shutdown Cooler

a. This equipment shall be operated to conform to the following standards:

1) The inlet gas temperature, as measured by TE-103, shall not exceed 1250°F.

2) The exit gas temperature, as measured by TI-111, shall not exceed 450°F.

3) Pressure in the system, as measured by PR-102, shall not exceed 15 psig.

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*S designates short form used as check-off sheet by fully trained operator.

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5724	Removal of Ruptured Fuel Element
5725	Operation of Reactor Temperature Monitor
5726	Operation Pitot-Static System
5727	Pre-Startup Check of "Water in Rods" Indicating System
5727-S	Pre-Startup Check of "Water in Rods" Indicating System
5728	Normal Operation of the Tube Sheet Cooling Water Pump
5729	Draining and Pressure Test of Reactor After Flooding
5731	Reactor Cover Replacement
5732	GCRE Facility Overnight and Weekend Shutdown
5732-S	GCRE Facility Overnight and Weekend Shutdown
5733	Routine Surveillance (Reactor Shutdown)
5734	RAM System Operation
5735	Operation of Reactor Instrumentation and Control Rod Back-up Pressure Systems
5735-S	Operation of Reactor Instrumentation and Control Rod Back-up Pressure Systems
5736	Fuel Element Orifice Replacement
5737	Control Rod Rack Removal
5738	Control Rod Rack Replacement
5739	Fuel Element and Pressure Tube Inspection
5801	Emergency Shutdown Procedures
5802	Scram, Setback and Annunciator Circuit Schedule
5803	Operation of Annunciator System
5805	Emergency Procedure - Power Failure - Scram
5806	Emergency Procedure - Main Loop - Low Flow Scram
5807	Emergency Procedure - Main Loop - High Temperature Scram
5808	Emergency Procedure - Power Level High Scram
5809	Emergency Procedure - Seismometer Tripped - Scram
5810	Emergency Procedure - Reactor Doors Open, Rev. 1
5811	Emergency Procedure - Run-Safe Switch - Safe Scram

ANSOP NO.	TITLE
5812	Emergency Procedure - Log-N High Level Scram
5813	Emergency Procedure - Period Short - Channel I, II, III, or IV - Scram
5814	Emergency Procedure - Period Short - Channel III or IV - Scram
5816	Emergency Procedure - Safety Rods - Low Pressure - Scram
5817	Emergency Procedure - Tube Bundle Cooling Pump Failure
5820	Emergency Procedure - Period Short - Channel I or II - Setback
5821	Emergency Procedure - Main Loop - High Temperature Setback
5822	Emergency Procedure - Main Loop - Low Flow Setback
5823	Emergency Procedure - Power Level High - Setback
5827	Emergency Procedure - Fuel Oil Supply - Low Pressure
5828	Emergency Procedure - Fuel Oil Storage Tank - Low Level
5829	Emergency Procedure - Utility Water Storage - Low Level
5830	Emergency Procedure - Utility Water System - Low Level
5831	Emergency Procedure - Utility Water Storage - High Level
5832	Emergency Procedure - Utility Air System - Low Pressure
5833	Emergency Procedure - Safety Rods - Pressure High or Low
5834	Emergency Procedure - Steam Boiler Trouble
5835	Emergency Procedure - Boiler Day Tank - High Level
5836	Emergency Procedure - S-201 Outlet - High Conductivity
5837	Emergency Procedure - S-301 Outlet - High Conductivity
5838	Emergency Procedure - Cooling Tower Basin, Low Level
5839	Emergency Procedure - Cooling Tower Basin, High Level
5840	Emergency Procedure - Emergency Power Supply Failure
5841	Emergency Procedure - Shutdown Cooling Loop - Low Flow
5842	Emergency Procedure - Shutdown Cooling Loop - High Temperature
5843	Emergency Procedure - Blowdown N_2 Supply - Low Pressure
5844	Emergency Procedure - Fuel Element - High Temperature
5845	Emergency Procedure - Main Compressor - Low Speed
5846	Emergency Procedure - Compressor or Gear Exit Oil High Temperature
5847	Emergency Procedure - Main Compressor - Lube Oil Low Pressure
5848	Emergency Procedure - Main Compressor Set Lube Oil or Seal Water Reservoirs - Low Level

ANSOP NO.	TITLE
5849	Emergency Procedure - Main Loop Makeup - High Flow
5850	Emergency Procedure - Main Loop - High Humidity
5852	Emergency Procedure - Main Loop - Oxygen Content High
5853	Emergency Procedure - Oxygen Storage - Low Pressure
5854	Emergency Procedure - Instrument Air - Low Pressure
5855	Emergency Procedure - Main Loop - Low Flow
5856	Emergency Procedure - Main Loop - High Pressure
5857	Emergency Procedure - Main Loop - Low Pressure
5858	Emergency Procedure - Main Loop - Temperature High
5859	Emergency Procedure - Demineralized Water System - Low Flow
5860	Emergency Procedure - Demineralized Water System - High Temperature
5861	Emergency Procedure - Cooling Water System - High Temperature
5862	Emergency Procedure - Cooling Water System - Low Pressure
5863	Emergency Procedure - Waste Nitrogen Storage Tank - High Level
5864	Emergency Procedure - Reactor Pool - Low Level
5865	Emergency Procedure - Reactor Pool, Level High
5866	Emergency Procedure - Demineralized Water Surge Drum Low Level
5867	Emergency Procedure - D-203 Waste Tank - High Level
5868	Emergency Procedure - D-202 Waste Tank - High Level
5869	Emergency Procedure - Demineralized Water Surge Drum - Low Pressure
5870	Emergency Procedure - Main Loop, E-101 - High Nitrogen Temperature
5871	Emergency Procedure - RAMS - High Activity
5872	Emergency Procedure - Nuclear Instrument Failure
5873	Emergency Procedure - Water in Main Loop
5874	Emergency Procedure - Thermocouple Manifold High or Low Pressure
5877	Emergency Procedure - Water in Rods
5878	Emergency Procedure - Main Compressor - Seal Water Low Pressure
5879	Emergency Procedure - Fission Product Monitor - High Activity
5880	Emergency Procedure - Gas in Fired Heater Pit

ANSOP NO.	TITLE
6001	Temporary Plant Shutdown Procedure
6001-S	Temporary Plant Shutdown Procedure
6160	Utility Water System, Normal Startup
6160-S	Utility Water System, Normal Startup
6170	Utility Water System, Normal Operation
6171	Domestic Water Bacteria Control
6180	Utility Water System, Normal Shutdown
6190	Utility Water System, Emergency Procedure
6260	Normal Startup, Cooling Water System
6270	Cooling Water (WC), Normal Operation
6271	Cooling Water pH Control
6272	Cooling Water Slime Control
6273	Cooling Water Chlorine Control
6274	Cooling Water Alkalinity Control
6275	Cooling Water Dianodic 130 Control
6276	Cooling Water - Blowdown Control
6280	Cooling Water System, Normal Shutdown
6290	Cooling Water System, Emergency Procedure
6360	Utility Air System, Normal Startup
6360-S	Utility Air System, Normal Startup
6370	Utility Air System, Normal Operation
6380	Utility Air System, Normal Shutdown
6380-S	Utility Air System, Normal Shutdown
6390	Utility Air System, Emergency Procedure
6460	Instrument Air System, Normal Startup
6460-S	Instrument Air System, Normal Startup
6461	Instrument Air Dryer, Normal Startup
6470	Instrument Air System, Normal Operation
6471	Instrument Air Dryer, Normal Operation
6480	Instrument Air System, Normal Shutdown
6480-S	Instrument Air System, Normal Shutdown
6481	Instrument Air Dryer, Normal Shutdown
6490	Instrument Air System, Emergency Procedure
6491	Instrument Air Dryer, Emergency Procedure
6510	Filling Oil Tank, T-302

ANSOP NO.	TITLE
6560	Fuel Oil System, Normal Startup
6570	Normal Operation , Fuel Oil System
6580	Fuel Oil System, Normal Shutdown
6590	Fuel Oil System, Emergency Procedures
6660	Liquid Petroleum Gas System, Normal Startup
6670	Liquid Petroleum Gas System, Normal Operation
6680	Liquid Petroleum Gas, Normal Shutdown
6690	Emergency Procedures, Liquid Petroleum Gas System
6710	Draining-Cleaning-Filling Steam Boiler
6760	Steam System, Normal Startup
6770	Steam System, Normal Operation
6780	Steam System, Normal Shutdown
6790	Steam System, Emergency Procedures
6810	Operation of the Demineralized Water Supply System
6810-S	Operation of the Demineralized Water Supply System
6820	D-201 Flooding Tank Fill Procedure
6860	Demineralized Water System, Normal Startup
6860-S	Demineralized Water System, Normal Startup
6870	Demineralized Water System, Normal Operation
6880	Demineralized Water System, Normal Shutdown
6880-S	Demineralized Water System, Normal Shutdown
6890	Demineralized Water System, Emergency Procedure
6950	Heating and Ventilating System Description
6960	Heating and Ventilating System, Normal Startup
6960-S	Heating and Ventilating System, Normal Startup
6970	Heating and Ventilating System, Normal Operation
6980	Heating and Ventilating System, Normal Shutdown
6980-S	Heating and Ventilating System, Normal Shutdown
6990	Heating and Ventilating System, Emergency Procedure
7000	Preventive Maintenance
7010	Welding and Cutting
7100	Modification to Class BB, D and E Piping
7101	Facility Modification

ANSOP NO.	TITLE
8000	Preventive Maintenance Procedures
8102	Facility Equipment Maintenance, Electrical
9100	Pre-Critical Testing - General
9102	Utility Water System, Check Out
9103	Utility Water System, Run In
9104	Cooling Water System, Check Out
9105	Cooling Water System, Run In
9106	Utility Air System, Check Out
9107	Utility Air System, Run In
9108	Instrument Air System, Check Out
9109	Instrument Air System, Run In
9110	Fuel Oil System, Check Out
9111	Fuel Oil System, Run In
9112	Liquid Petroleum Gas, Check Out
9113	Liquid Petroleum Gas, Run In
9114	Steam System, Check Out
9115	Steam System, Run In
9116	Demineralized Water System, Check Out
9117	Demineralized Water System, Run In
9118	Heating and Ventilating System, Check Out
9119	Heating and Ventilating System, Run In
9120	Fresh and Waste Nitrogen Systems, Check Out
9121	Fresh and Waste Nitrogen Systems, Run In
9123	Oxygen System, Run In
9124	Waste Water System, Check Out
9125	Waste Water System, Run In
9126	Chemical Wash System, Check Out
9127	Chemical Wash System, Run In
9128	Pool Water System, Check Out
9129	Pool Water System, Run In
9130	Electrical System, Check Out
9131	Lube Oil System (OL), Check Out
9132	Lube Oil System (OL), Run In
9133	Initial Filling of the Reactor Pool

ANSOP NO.	TITLE
9134	Fired Heater, FH-101, Check Out
9135	Seal Water System, Check Out
9136	Seal Water System, Run In
9137	Bridge Crane, Check Out
9138	Bridge Crane, Run In
9139	Nitrogen Gas Dryer, Check Out
9140	Nitrogen Gas Dryer, Run In
9141	Main Compressor, Check Out Acceptance Procedure
9142	Main Compressor, Run In Acceptance Procedure
9143	Emergency Power System, Check Out
9144	Emergency Power System, Run In
9145	Gas Analyzer and Control System, Check Out and Tuning
9147	Instrument Air Dryer, Check Out
9148	Instrument Air Dryer, Run In
9149	Cooling Water System, Pretreatment
9150	Microsen Controller Tests and Adjustments
9152	Demineralized Water Supply System, Run In
9153	Foam System, Check Out and Run In
9155	Chlorinator, Check Out
9156	Chlorinator, Run In
9157	Water Softener, System Run In
9158	Water Softener System, Normal Operation
9159	Water Hardness Test
9200	Initial Critical Experiment
9201	Full Core Flooded Test
9202	Dry Critical Experiment
9203	Operating Core Experiment
9204	Reactor Drying Test
9207	Initial Critical Experiment-II
9208	Moderator Temperature Coefficient Determination
9209	Core Power Mapping Experiment
9210	Power Ascension
9211	Fuel Element Storage Rack Safety Test
9212	Core Heat Loss Experiment
9220	Insulation Depressurization Test

ANSOP NO.	TITLE
9221	Main Loop Leakage Test
9223	Main Loop High Pressure Proof Test
9225	Main Loop High Temperature Proof Test at Low Pressure
9226	Control Rod Performance Test
9227	Shutdown Cooling System Test
9229	Main Loop High Temperature Proof Test at Operating Pressure
9230	Emergency Cooling Systems Performance Test
9232	Reactor Flooding Test
9240	Main Loop, External Portion , Leakage Test
9242	Main Loop, External Portion, High Pressure Proof Test
9243	Main Loop, External Portion, High Temperature Proof Test
9245	Main Heat Dump Systems Transient Tests
9300	Radiological and Chemical Analyses
9301	Wire Activity Counting
9302	Fission Burnup by CS-137 Determination
9323	P_h Determination
9324	Conductivity Determination for Water
9325	Oil in Water Analysis
9330	Flash Point Test of Fuel Oil
9331	Total Solids in Water
9332	Gross Activity in Aqueous Solution
9333	Chlorine in Water
9334	Dianodic Determination in Cooling Water Analysis
9335	Cycles of Concentration for Cooling Water
9363	Nitrogen Coolant Gas Sampling
9364	Dewpoint Determination of Nitrogen or Instrument Air
9365	Oxygen Content of Nitrogen Gas
9372	The Determination of CO_2, O_2, and Co in Combustion Gases
9402	Core Flux Mapping with the IB-9ØT Element
9500	Initial Critical Experiment with the Stainless Steel Tube Bundle
9501	Full Core Flooded Test with the Stainless Steel Tube Bundle
9502	Dry Critical Experiment with the Stainless Steel Tube Bundle
9503	Operating Core Experiment with the Stainless Steel Tube Bundle
9504	Reactor Drying Test - II

ANSOP NO.	TITLE
9505	Low Power Neutron Instrument Calibration
9506	Reactivity Worth Experiment
9509	Core Flux Mapping Experiment
9600	Wet Critical Experiment with IB-2L Elements
9601	Full Core Flooded Test with the IB-2L Elements
9602	Dry Critical Experiment with the IB-2L Fuel Elements
9603	Operating Core Experiment with the IB-2L Fuel Elements
9604	Emergency Shutdown Cooling Experiment
9605	Reactivity Coefficients
9606	Reactivity Worth Experiment
9607	Reactivity Coefficients - II
9609	IB-2L Core Flux Distribution Measurement
9610	Power Ascension with the IB-2L Fuel Elements
9612	Subcriticality Proof Test of the Aluminum Tube Storage Rack
9630A	Reactivity Coefficients - III
9631	Fuel Element Reactivity Determination

APPENDIX H

GCRE-I TRAINING COURSE CURRICULUM

Course No.	Title	Number of Sessions
1	Elements of Supervision	7
2	First Aid (by Red Cross)	-
3	Process Training	8
4	Radiological Safety	5
5	Administrative Procedures	3
6	Reactor Operation	7
7	Experimental Procedures	2
8	Safety	5
9	Security	2

Course 1

ELEMENTS OF SUPERVISION

SESSION 1 Safety

1. The supervisor's responsibility for safety
2. Setting examples for employees
3. Discussion and practice problems

SESSION 2 Maintaining Discipline

1. Forcefulness
2. Sternness
3. Familiarity
4. Respect
5. Confidence

SESSION 3 Handling Grievances

SESSION 4 Correspondence and Communications

SESSION 5 Work Direction

1. Leader - follower
2. Teamwork

SESSION 6 Disciplinary Action

1. Reprimands

SESSION 7 Training

1. New men
2. Self-training

Course 3

PROCESS TRAINING

SESSION 1 Process Instrumentation (General)

1. Pressure switches
2. Transmitters
3. Recorders
4. Controllers
5. Flow meters
6. Motor valves
7. Relief valves
8. Annunciator station

SESSION 2 Process Instruments (Special)

1. Main flow meter
2. BTUR
3. Tachometer
4. Remote set point equipment
5. Protect-o-relay
6. HIC-OIC
7. Alarm systems (ADT, etc.)
8. Conductivity meters
9. RAMS

SESSION 3 Equipment (General-1)

1. Pumps
2. Heat exchangers
3. Demineralizers
4. Dryers

SESSION 4 Equipment (General-2)

1. Filters
2. Heating and ventilating
3. Air compressors
4. Chlorinator and hypochlorinator

SESSION 5 Oil-Fired Heater, Cooling Tower and Emergency and Shutdown Cooling

SESSION 6 Compressor and Auxiliaries

SESSION 7 Heat Removal Systems Operation

SESSION 8 Auxiliary Systems Operation

Course 4

RADIOLOGICAL SAFETY TRAINING PROGRAM

SESSION 1 Fundamental Radiation Physics

1. Nature of matter
2. Properties of matter
3. Radioactivity
    a. Definition
    b. Historical background
        (1) Becquerel
        (2) Curies
        (3) Rutherford
        (4) Chadwick
    c. General Properties
        (1) Half-life
        (2) Units
4. Properties of Elementary Particles
    a. Proton
        (1) Mass
        (2) Charge
        (3) Range
    b. Beta
        (1) Mass
        (2) Charge
        (3) Range

c. Alpha

(1) Mass
(2) Charge
(3) Range

d. Neutron

(1) Mass
(2) Charge
(3) Range

e. X-Gamma

(1) Mass
(2) Charge
(3) Range

5. Nuclear Fission and Fission Products

a. Nuclear Reaction

(1) Energy Release
(2) Chain Reaction

b. Fission Products

(1) Decay

NOTE: Visual Aid - Films (1) "Atomic Energy" - 10 minutes
(2) "Unlocking the Atom" - 20 minutes

SESSION 2 Biological Effects of Radiation

1. Radiological Units

a. Dose - Dose Rates

(1) Roentgen
(2) Rem
(3) Rep, etc.

2. Ionization in Tissue

3. Relative Biological Effectiveness (RBE)

a. Table

4. External Radiation Effects

   a. Body Organs

      (1) α
      (2) β
      (3) x-γ
      (4) η

   b. Skin

      (1) α
      (2) β
      (3) x-γ

   c. Symptoms of Overexposures

      (1) Table

5. Internal Radiation Effects

   a. Whole Body

   b. Partial (localized deposition)

   c. Effective Half-life

   d. MPC - Maximum Permissible Concentration

NOTE: Visual Aid - Film "Atomic Biology for Medicine" - 12 minutes

SESSION 3 Dosimetry and Radiation Measurements

1. Personnel Dosimetry
2. Gamma Detector
3. Beta Detector
4. Alpha Detector
5. Neutron Detector
6. Laboratory Counting Equipment
7. Radioactive Aerosol Detection and Measurements

SESSION 4 Basic Radiological Safety

1. Introduction

a. Mental Attitude

b. Know-how

2. Radiological Hazard Control

a. External Exposure Control

(1) Time

(2) Distance

(3) Shielding

(4) Decay

b. Internal Exposure Control

(1) Protective Equipment

(2) Aseptic Habits

c. Contamination Control

(1) Protective Clothing

(2) Change Room and Monitoring Stations

(3) Personal Survey

d. RCZ - Radiation Control Zone

e. SWP - Special Work Permit

3. H.P. Calculations

a. Dose

(1) R = 6CE

(2) Inverse Square Law

b. Stay Time

c. Shielding

(1) Tenth Value

(2) $I = B(x)I_o e^{-\mu x}$

4. Radiation Monitoring

a. Evaluation

b. Techniques

c. β, γ Survey

d. α Survey

e. η Survey

f. Removable Contamination Survey

g. Air

h. Water

i. Gas

j. Clothing

k. Record Keeping

5. Decontamination

a. Area

b. Equipment

c. Personal

6. Radioactive Waste Disposal

a. Packaging

b. NRTS Standards

7. Radioactive Storage

a. Container

b. Tagging

c. Shielding

NOTE: Films (1) "Living with Radiation" - 28 minutes
(2) "Protecting the Atomic Worker" - 12-1/2 minutes
(3) "Primer on Monitoring" - 30 minutes

Course 5

ADMINISTRATIVE PROCEDURES

SESSION 1 Industrial Relations

1. Management Policy
2. Employee Services
    a. Group Insurance Program
    b. Retirement and Pension Program
    c. Length of Service Awards
    d. Welfare and Recreation Club
    e. Stock Purchase Plan
    f. Bulletin Boards
    g. Medical
3. Sick Leave Policy
    a. Hourly
    b. Salaried
4. Vacation Policy
    a. Hourly
    b. Salaried
5. Holidays
6. Rules of Conduct

SESSION 2 Accounting and Purchasing

1. Time Cards
    a. Preparation
    b. Authorization
2. Payroll
    a. Pay Policy

3. Procurement

   a. Phillips Stores

   b. AGN Purchases

   c. Petty Cash Purchases

4. Shipping and Receiving

SESSION 3 <u>Idaho Relocation Policy and General</u>

1. Expense and Pay Policy for Employees Assigned to NRTS, Idaho Falls, Idaho

   a. Employee and Family Travel

   b. Household Goods

   c. Per Diem for Locating Period

2. General

   a. Travel Orders

   b. Expense Reports

   c. Reproduction Services

   d. Photographic Services

   e. Use of Government Vehicles

   f. Use of Company Car

   g. Long Distance Telephone Calls

   h. Office Letters

   i. Typing Procedures

   j. Bus Transportation

Course 6
REACTOR OPERATION

SESSION 1 Fundamentals

1. Atomic Structure
   a. Atomic Number and Mass Number
   b. Isotopes
2. Atomic Particles
   a. Electrons, Positrons
   b. Protons, Deuterons
   c. Alpha Particles
   d. Neutrons
3. Nuclear Reactions
   a. Scattering
   b. Capture
   c. Fission
4. Cross Section

SESSION 2 Reactor Theory

1. Basic Reaction
2. Multiplication Factor
3. Critical Size
4. Thermalization
5. Energy Removal
6. Shielding

SESSION 3 Reactor Kinetics

1. Reactivity
2. Reactor Periods

3. Temperature Coefficients
4. Fission Product Poisoning

SESSION 4 Instrumentation

1. Fission Chambers
2. Ionization Chambers
   a. Compensated
   b. Uncompensated
3. Count Rate Systems
4. Period Systems
5. Log N and Linear Level Systems

SESSION 5 Reactor Control

1. Methods of Control
2. Types of Control Rods
3. Control Rod Operation
4. Safety Systems
5. Automatic Control Systems

SESSION 6 GCRE-I

1. Instrumentation
2. Control
3. Heat Removal
4. Shielding

SESSION 7 Reactor Operation

1. Initial Critical Experiment
2. Normal Startup
3. Normal Operation
4. Normal Shutdown
5. Emergency Shutdowns
6. Recovery From Emergency Shutdowns

Course 7

EXPERIMENTAL PROCEDURES

SESSION 1 Zero Power Experiments

1. General Remarks
2. Initial Critical Experiments
3. Rod Worth Determinations
4. Coefficient Determinations
5. Flux Mapping

SESSION 2 System Transient Experiments and Fuel Element Testing

1. General Remarks
2. Nitrogen Coefficient
3. Reactivity Dependence with Flow and Temperature
4. Fuel Element Testing

Course 8

SAFETY

SESSION 1 General

1. Philosophy of Industrial Safety Program
   a. Individual Responsibility
   b. Company Responsibility
   c. Contractual Responsibility
2. GCRE-I Safety Program
   a. Central Safety Committee
   b. Safety Program Committee
   c. Investigation Committee
   d. Safety Rules and Procedures Committee
   e. Inspections
   f. Meetings
   g. Training
   h. Drills
   i. Safety Rules
3. Rules of Conduct
   a. Running, Horseplay, etc.
   b. Clothing

SESSION 2 Mechanical Safety

1. Types of Safety Hazards
   a. Moving Machinery
   b. Tripping
   c. Bumping
   d. Slipping and Falling

2. Safeguards

   a. Guards

   b. Isolation

   c. Safety Rules

   d. Safety Training

3. Discussion of GCRE-I Facility Hazards

SESSION 3 Lock and Tag Procedure

SESSION 4 Closed Vessel Entry Procedure

SESSION 5 Safety and Housekeeping Inspections

Course 9

SECURITY

SESSION 1 Physical Security

1. Physical Security Standards

   a. Definitions
   b. Security Areas
   c. Physical Barriers
   d. Personnel Identification
   e. Protective Alarms
   f. Guard Force
   g. Protective Communications
   h. Security of Matter in Storage

2. Pass and Badge Procedure

   a. Requirements for Limited and Exclusion Area
   b. Types of Identification
   c. Lost or Forgotten Identification Media
   d. Use or Misuse of Identification Media

3. Employee Start and Termination Procedure

   a. Security Acknowledgement
   b. Security Termination Statement

4. Personnel Security Clearance

   a. Policy
   b. Types

5. Control of Visits

   a. Classified Visits
   b. Unclassified Visits

6. Security Education
7. GCRE-I Restricted Area

SESSION 2 Document Control

1. Document Control Center
   a. Custodian
   b. Location
   c. Records
2. Control of Classified Information
   a. Types of Classified Information
   b. Classification Categories
3. Preparation, Documentation, and Transmittal of Classified Documents
   a. Preparation
   b. Chargeouts and Transfers
4. Loss or Unaccounted for Classified Documents
   a. Reporting Requirements
   b. Disciplinary Action
5. Security Infractions

Report No. IDO-28598

L E G A L   N O T I C E

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The information herein is regarded as preliminary and subject to further checking, verification and analysis.

NLC 0891
NAA-SR-8888
ANL-6761
UCRL-1088
SCR-610
IDO-16907

GENERAL TIRE

AEROJET-GENERAL NUCLEONICS